NUCLEAR SAFETY

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U. S. ATOMIC ENERGY COMMISSION, by OAK RIDGE NATIONAL LABORATORY

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NUMBER 1

TECHNICAL PROGRESS REVIEWS

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NUCLEAR SAFETY

A REVIEW OF RECENT DEVELOPMENTS

Prepared for DIVISION of TECHNICAL INFORMATION,
U. S. ATOMIC ENERGY COMMISSION,
by OAK RIDGE NATIONAL LABORATORY

● SEPT. 1961

● VOLUME 3

● NUMBER 1

foreword

This quarterly journal is a Technical Progress Review prepared by the Oak Ridge National Laboratory at the request of the Division of Technical Information, U. S. Atomic Energy Commission. This Review is intended to assist those interested in keeping abreast of significant developments in the field of nuclear safety. *Nuclear Safety* is not a comprehensive abstract of all literature published in this field during a given quarter; rather it is a mechanism for presenting concise reviews of selected subjects as prevailing interest and available information warrant.

Coverage of the Review is limited to topics relevant to the analysis and control of hazards associated with nuclear reactors, operations involving fissionable materials, and the products of nuclear fission. Primary emphasis is on safety in reactor design, construction, and operation; however, safety considerations in reactor fuel fabrication, spent-fuel processing, nuclear waste disposal, and related operations are also treated. Safety in the use of radioisotopes in industry, medicine, and research is excluded, as are most topics considered to be the province of health physics. Even with these exclusions, nuclear safety overlaps such diverse fields as nuclear physics, solid-state physics, mechanics, chemistry, meteorology, geology, seismology, metallurgy, law, and nearly all branches of engineering. The authors will therefore review material from these fields which, in their opinion, has a direct bearing on nuclear safety.

Two distinctly different types of articles are in this issue of *Nuclear Safety:* reviews of current literature and special review articles on specific topics. The editors feel that each of these articles makes a necessary and distinctive contribution to this journal. The special review articles permit discussion of pertinent subjects which cannot be adequately considered by reference to only the current literature. The current review articles, however, constitute the major portion of this issue. All incoming literature (including reports, books, American and foreign technical journals, and transactions) is examined for subjects within our area of interest. This material is collected, grouped, and reviewed by experts. Interpretations, in any article, represent the opinions of the editors, who are employees of the Oak Ridge National Laboratory. Readers are urged to consult the references to original work for more complete information.

It is recognized that the critical evaluation of subject areas leading to the determination of criteria cannot fail to stimulate contrary opinions. This is expected to be particularly true in the area of nuclear safety since, in many instances, only preliminary information is available, the ramifications are many and varied, and opinion and judgment must be relied upon so heavily. Although the editors do not propose that *Nuclear Safety* act as a clearing house for safety correspondence because of the above facts, we have had for some time a policy that would permit the publication of statements of position at variance with those expressed by the editors. Such statements will be published after the editors have ascertained that a real difference exists and that the position is reasonable.

In addition to the invited contributors, many members of the Oak Ridge National Laboratory staff wrote review material, reviewed manuscripts, or otherwise contributed to this publication. Their contributions are gratefully acknowledged.

W. B. COTTRELL, Editor; R. A. CHARPIE, Advisory Editor
C. G. BELL, E. E. GROSS, W. de LAGUNA, L. A. MANN, A. W. SAVOLAINEN, and C. S. WALKER, Assistant Editors
Oak Ridge National Laboratory

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Fire

By F. I

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Section I

GENERAL SAFETY CRITERIA

*Fire-Protection Considerations in Nuclear Incidents**

By F. L. BRANNIGAN†

The fire department's interest in the so-called "criticality" or nuclear accident arises from the number and variety of ways in which normal fire-service practice must be modified where the possibility of such incidents exists. These fire-service problems were discussed in a U. S. Atomic Energy Commission (AEC) publication entitled *Living with Radiation.*[1]

Although a criticality accident might occur which would not involve the fire department, the fire department could be involved in several ways. For instance, a fire or an explosion might result from a criticality accident or happen simultaneously; a fire might cause a criticality accident; and the fire department might be responsible for providing rescue, ambulance, and decontamination service for victims of an accident. It is of interest to note that the first indication of the Stationary Low-Power Plant No. 1 (SL-1) incident was received through an automatic fire-alarm system.

In this article no consideration is given to the unique problems posed by the combustion or oxidation of reactor core constituents *in situ.* Such credible fires as the burning of reactor graphite (as at Windscale) or combustion of fuel elements and/or their cladding should be anticipated in advance in the reactor design. The facility design must be such as to prevent the spread of such fires and otherwise provide for their control. The control of such fires would not normally be a function of the community fire service, although as a constituent of any disaster organization, the fire service might participate.

Up to this writing, no criticality accident has been caused by, or has caused, a fire, but six people have been killed in criticality accidents, including three in the recent SL-1 accident. All six of these victims were immediately adjacent to the critical mass. Personnel as close as 8 ft have survived.

The fact that water reflects and moderates neutrons raises serious questions with respect to fire-protection and fire-fighting procedures. In prefire planning where critical-mass quantities of fissionable materials are involved, the problems of possibly disparate hazards must

*The views expressed in this article do not necessarily represent those of the U. S. Atomic Energy Commission or its appropriate office or division.

†Francis L. Brannigan is a nuclear safety training specialist with the Safety and Fire Protection Branch of the AEC Office of Health and Safety, and he brings to this position many years of experience and training in safety and fire-prevention work. He was born and raised in New York City, and he received the B.B.A. degree from Manhattan College in 1938. During World War II he was Officer in Charge, U. S. Naval Fire Fighting School, Balboa, C. Z.; and, in the early postwar years, he organized the Navy's first shore station Fire Protection Training Facility. In 1948, Brannigan became Chief of the Safety and Fire Protection Branch of the AEC New York Operations Office. In his present capacity as training specialist, he has spent a number of years in the development of programs for the training of laymen in a sound appreciation of radiation hazards. For his work in this area, he received a citation in 1957 from the Training Officers Conference and a Superior Performance Award from the AEC in 1959. Brannigan edits the publication, *The Burning Question,* a newsletter on fire protection, distributed by the National Catholic Welfare Conference, and is one of the authors of the government publications *Living with Radiation* and *Radiation Safety Primer.* He is a member of the Society of Fire Protection Engineers and the Baltimore-Washington chapter of the Health Physics Society.

be considered; that is, the fact that water can decrease factors of nuclear safety where fissionable materials are involved is diametrically opposed to the fact that water is the only practical extinguishing medium for large volumes of ordinary combustible materials. Further, the procedures for providing rescue and ambulance service for casualties must take into account the possibility of the presence of intense radiation.

Use of Water in Fire Fighting

Since water is a reflector and moderator of neutrons, the indiscriminate use of water must obviously be avoided in fire-fighting situations where fissionable materials are being handled. On the other hand, there has been a tendency to prohibit the use, and even the supply, of water without actually examining the possible criticality hazards. The result of this has been that, in several serious fires, the use of water was delayed until an estimate of the situation could be made and a decision rendered that the use of water would not bring about a criticality accident. In addition to the criticality hazard, water provides a convenient vehicle for the spread of contamination. Therefore its use must also be accompanied with due consideration for the subsequent contamination and disposition.

Since the calculation of the possibility of a criticality accident requires a high order of mathematical competence and necessarily takes time, the need for a preexamination of the situation by persons who are expert in the field of criticality is strongly indicated. These studies should be directed not only to storage, shipment, and use of fissionable material but also to criticality problems associated with the use of water on fissionable material. Such studies should assume that the fissionable materials are in the worst possible geometry and are flooded with water. Although the potential hazards of such an assemblage would have to be evaluated by an expert, a number of useful criticality guides are available (see the article entitled "Developments in the Safe Handling of Fissionable Materials," page 4 of this issue). In addition, the National Fire Protection Association has issued a report[2] on fire-protection standards for laboratories that handle radioisotopes.

Although facilities for radioactive and nuclear materials would normally be expected to be of fireproof construction, if the building is of combustible construction or of noncombustible construction with a heavy loading of combustible contents, the only extinguishing medium worth considering is water. (It is indeed fortunate that water is usually readily available in large quantities at relatively low cost. Other extinguishing agents have been developed for specialized use, but they are costly. Further, the equipment required for their storage and use is expensive both in terms of materials and space.) It must therefore be assumed that water will be used in the control of the fire. If an analysis shows that the use of water could bring about a significant decrease in the criticality safety of the situation, a choice of two procedures must be made. The fire can be allowed to burn out without the application of water; however, this is practical only at isolated locations where the building, the contents, and the entire surrounding area are under control and the management is willing to accept this approach to the problem. Otherwise, either the criticality hazard or the combustible loading must be adjusted to the situation. In other words, either the fissionable material must be handled in a fire-resistant building without combustible contents, in which case the need for water in fire fighting can be essentially ignored, or, if the combustible contents or combustible building cannot be avoided, then the fissionable materials must be handled, stored, and used in ways that will prevent a criticality accident as a result of the use of water. It is simply not sufficient to dismiss the problem by saying, "we will use no water," if, in fact, a combustible loading exists which could require the use of water.

In the shipment of fissionable materials, a very realistic approach has been taken to the problem of water flooding. It is presumed that any shipment of fissionable materials might be totally submerged in water and then taken out of the water. In this event all the containers would be wet, and all openings, crevices, etc., would retain water. The shipping containers are therefore designed so that, even under these conditions, criticality cannot occur. The restrictions on the shipment of fissionable materials have been set forth in AEC regulations,[3] and the shipment of fuel elements, in particular, was discussed in a previous issue of *Nuclear Safety*.[4]

Both plutonium and uranium metal will burn readily if exposed to air in finely divided form,

and even sizable solid pieces have ignited. However, the handling of a "natural" or "depleted" uranium metal fire is fairly simple because water can be used. Although hydrogen gas is evolved, there are no violent reactions. Automatic sprinklers have been found to be quite suitable for protection of normal and depleted uranium scrap.[5] Automatic sprinklers have also been installed in buildings processing fissionable materials. A study was made of sprinkler discharge vs. criticality in connection with the fire protection of gaseous diffusion plants.[6]

Since finely divided natural-uranium metal may ignite spontaneously if not stored under oil or water, it is apparent that enriched uranium in the same physical form may be just as readily ignited, since enriched uranium and natural uranium are chemically identical. However, because of the reflecting and moderating characteristics of water and the consequent criticality hazard, it would not be possible to use water indiscriminately on a fire involving enriched uranium. On the other hand, the criticality hazard greatly limits the fire-fighting problem because criticality limitations restrict the amount of material in any container to a very small quantity, and the distance between containers required to prevent criticality accidents is such that there would be little probability of extension of fire from container to container.

A fire involving enriched material would usually be allowed to burn out. The burnout should, if possible, take place in an air-collecting system so that the off-gases can be recovered and contamination can be controlled. The use of special fire-extinguishing agents, such as salt or graphite, is not generally recommended because, if the metal is allowed to burn freely, it will go to the oxide, and, at some expense, the oxide can be returned to its original form. When fire-fighting agents are mixed with the enriched metal, it is usually impossible to recover the pure metal, and the loss may be much higher than if the fire were allowed to burn.

Rescue Service

To date, six people have died in criticality accidents, and several others have received high radiation exposures. Three of the individuals who died were able to leave the area of the accident under their own power. However, in the accident at Los Alamos in December 1958, the man who died received an exposure of several thousand roentgens equivalent man (rem) and became irrational shortly after the accident. This raises the possibility that a person who has received a high dose of radiation might become so irrational and confused that he would remain in the area and make no attempt to leave. There is also the possibility that some other circumstances surrounding the accident might make it impossible for the man to leave. He might be injured or trapped under damaged equipment, etc. A question naturally arises as to whether an attempt should be made to rescue that individual. No specific guidance for making this decision can be given because each situation may obviously differ radically. General criteria for analysis of a specific incident can be developed, however, through cooperative study by members of the fire department and the technical staff. Preevaluations of the situations that might exist after a criticality accident could provide a basis for determining whether rescue work would be feasible.

The situations that might be faced by rescue personnel after a criticality accident can be broken down into several broad categories. In one type of situation, the overexposed personnel have left the area under their own power, and the fire department simply has the responsibility of transporting them to the hospital. The only first-aid measure to be taken is to shower the individuals and to remove their clothes in order to remove any fission-product contamination from their bodies. The ambulance may become contaminated and should not be placed back in service until it has been checked.

In a second type of situation, the criticality accident may occur simultaneously with some other accident, and the personnel who have been exposed are trapped. If the nature of the fissionable materials is such that the recurrence of a criticality accident is extremely improbable, rescue operations can be carried out. The rescue personnel should, of course, be masked to minimize inhalation of radioactive fission products. There may also be a high gamma-radiation level from fission products. In most instances the fire department would have the assistance of the local health physicist to measure radiation levels. Radiation safety training is available to all fire departments, and a manual on radiation and monitoring fundamentals for the fire service has been prepared.[7] The number of minutes that can be spent in rescue

operations can be determined by dividing the radiation rate in roentgens per minute into the radiation dose that the fire chief has determined he is willing to have his men accept under these circumstances.

A criticality accident involving a homogeneous solution would present a third type of situation. In this case, as mentioned above, the entry of other human bodies into the area might conceivably be sufficient to bring about another criticality incident. As a matter of fact, successive criticalities could occur. Under such circumstances, it might not be justifiable to risk lives in an attempt to rescue a man who has little chance of survival.

Obviously, the specific conditions cannot be predicted in advance of an accident. The fire department and the plant technical staff should, however, discuss the possible problems so that, if an accident occurs, the best decisions can be made under the circumstances, based on as much practical preplanning as possible.

Developments in the Safe Handling of Fissionable Materials

The general field of nuclear safety in fuel-processing facilities has two major aspects. The first of these is the development of technical standards or criteria that delimit the conditions under which fissionable material may be safely handled. The second phase encompasses the administrative methods for ensuring that these standards will be adhered to. Freedom of action in these fields, primarily the second one, is restrained somewhat by the regulations of governmental bodies having jurisdiction over various phases of the endeavor and the activities of national and international standardization groups. Some of the international standardization groups appear to have quasi-governmental status in several countries, but not in the United States.

This review discusses developments in the handling of fissionable materials subsequent to the previous reviews of this subject in *Nuclear Safety*.[8,9] Reactor considerations, except as they also involve other aspects of nuclear safety, are not separately discussed here. Similarly, no reference is made to the rather considerable effort, both administrative and technical, which has gone into the various aspects of handling criticality accidents.

Basic Criticality Data

The current theoretical work in the field of nuclear safety has involved, primarily, evaluations of experimental data rather than the development of completely new methods or approaches. A major point of interest has been the progress in the incorporation of criticality data into machine codes and in the application to criticality safety problems of some of the codes developed for reactor studies.[10–17]

This approach has included the use of multigroup calculations and has been valuable both for extending general criteria and for solving specific problems, including some which require multidimensional treatment. Such codes are periodically listed in the publication *Communication of the Association for Computing Machinery* and have been included under the general title of "Abstracts—Nuclear Reactor Codes." Some of the smaller data-processing machines have been used in the manner of desk calculators for rapid, repetitive calculations.

A major experimental development has been the work done at Hanford Atomic Products Operation (HAPO),[11] which has shown that the minimum, critical U^{235} enrichment for salts and solutions is about 1 per cent. This finding has permitted a basic change in the criticality criterion, as discussed below. The measurements were made with the Physical Constants Testing Reactor (PCTR), which does not give direct critical data;[11] however, confidence in the validity of these ingenious indirect measurements was enhanced by the agreement of data obtained by this method for material of 2 per cent U^{235} enrichment with data obtained for the same material in critical experiments at the Oak Ridge National Laboratory (ORNL).[12] Additional studies of the conditions for criticality at a U^{235} enrichment of 2 per cent are under way, and the measurements that permit the determination of minimum values for the various parameters involved will be of particular interest.[12] Interaction experiments have been performed with comparatively large arrays of small metal slabs and with cylinders containing uranium in solution.[15] The experiments with solutions are continuing. Additional data have been obtained in research facilities, although these experiments are usually directly applicable only to specific operational problems and are thus of more limited usefulness.

The published, unclassified, experimental data involving plutonium or U^{233} which have general applicability have shown the need for reevaluation of some of the minimum permissible values of the critical parameters.[16,17] In addition, it is expected that the plutonium research facilities at Hanford will provide additional pertinent data in the not-too-distant future.

Criteria and Administrative Methods

The limiting U^{235} enrichment of 0.72 per cent for salts and solutions which was used as a basis for determining the need for taking steps to prevent criticality was recently increased to 0.90 to 0.95 per cent, based on the finding that the minimum U^{235} enrichment required for criticality is approximately 1 per cent. The actual value used depends on the specific conditions and the degree of uncertainty in the enrichment determination. This change in the criticality criterion has resulted in only a negligible change in administrative methods where enriched uranium is used, but it has had a widespread effect on the number of facilities and locations where criticality considerations are required. The fact that the previous value was somewhat conservative indicates that adequate safety factors had been applied in developing criteria where technical information was incomplete. Minor changes in other criteria have also been based on new experimental data and reevaluations of the safety factors used. Further, the nuclear differences in alpha- and delta-phase plutonium as they concern nuclear safety are also being more widely applied.

The extension of experimental and theoretical information has permitted a better definition of critical conditions, and this has, in turn, led to increased use of criteria involving simultaneous control of several parameters, such as a combination of geometry, reflection, and moderation. Such control methods are appropriate and can be safely used, but they must be based on a more rigorous analysis than the control of a single factor. In order to ensure that the over-all factor of safety is adequate, the possible range of each of the variables must be considered individually and in combination with all the other pertinent variables. This multiparameter analysis takes into consideration the competing effects of the various control factors affecting criticality, but can be a special problem to those not intimately familiar with their use.

A significant change in administrative criteria has been the additional emphasis now being given to the use of nuclear poisons. The practical advantages of such poisons, either built into the equipment or incorporated as a part of the fissionable material, have long been recognized, but the administrative difficulties inherent in ensuring their constant presence have been a major deterrent to their more widespread use. In meeting such problems as the storage of large quantities of uranium solutions, however, fixed poisons have been used in the storage containers in some facilities. Although the precautions currently maintained in the use of poisons are probably adequate, it must be emphasized that nuclear poisons should be used cautiously and should be depended upon only if other safety methods are not feasible.

Nuclear Safety Guides and Other Literature

The basic nuclear safety documents,[18–21] which were previously reviewed,[8,9] provide summaries of information and administrative methods which have been found to be useful in nuclear safety work. A noteworthy recent addition is a United Kingdom Atomic Energy Authority (UKAEA) publication[22] that provides curves of minimum parameter values for critical conditions. The curves were developed from experimental and theoretical studies. Although "safe" data are not specifically listed, some suggestions for administrative control are included. In general, the UKAEA data quoted are similar to those used in the United States;[13] however, the curves of permissible U^{235} concentrations given for salts and solutions at U^{235} enrichments below about 5 per cent are, in some cases, significantly lower than the corresponding curves which had been previously determined in the United States[13] but which were not widely published until after the issuance of the UKAEA document. In addition, the assumption is made for the United Kingdom that the maximum uranium density of nonmetallic fuel systems can be 5.1 g/ml, instead of the 3.2 g/ml generally used in the United States. Although it is possible to produce high-density uranium oxide hydrates, they are not thought to be widely available in nature. Consideration of high-density hydrates may have had a considerable

influence on the United Kingdom figures, especially since no additional experimental information appears to be available to support their lower values. The data given for these conditions do not therefore appear to indicate that the corresponding values in the United States documents are unsafe for continuing operations.

Another rather complete document on the general subject of nuclear safety is the Eurochemic (European Company for the Chemical Processing of Irradiated Fuels) criticality guide.[23] This report not only covers the field rather thoroughly but it also gives examples for solving problems and has an excellent bibliography. It appears to follow standard criticality control methods rather closely and is an excellent addition to the field.

Criticality control is the subject of one chapter of a McGraw-Hill handbook on radiation hygiene[24] and provides the topic for several chapters of a Pergamon Press publication.[25]

Another document of some interest is a Convair report[26] that is really an example of a nuclear hazards evaluation of a particular type of operation, the fuel-fabrication cycle. It has apparently been prepared as a review of licensee operations in the United States, much in the manner required for reactors by the Advisory Committee on Reactor Safeguards (ACRS). The information contained in this document is generally adequate; however, reference to the more standard documents or experimentation is advisable for analyzing analogous, but not identical, processes and equipment.

A significant addition to this field is the newly revised Nuclear Safety Guide,[18] which has been generally updated. Graphs are incorporated to express safe conditions. As in the previous version, critical parameter values are not given. Also, as in the previous version, the document appears to be most useful to those already somewhat familiar with the field, especially in the administrative aspects of practical nuclear safety. For example, the inclusion of graphs provides a wider range of information concerning safe systems than has hitherto been available in a single document. Consequently, because interpolation and extrapolation may be involved, it becomes even more important that anyone designing or operating equipment wherein fissionable materials are involved be aware of the number and complexity of the many factors that can affect nuclear safety so that he will be able to ascertain that conditions of normal and readily possible abnormal operation cannot result in the predetermined limits of safety being exceeded, especially if the concurrent use of several parameters is involved. It may also be noted that the bases for some of the criteria given have not been published hitherto or otherwise made readily available and that some of the values given, although probably entirely valid, do appear to represent rather extreme extrapolations from experimental data.

Two other documents, the second of which is in preparation, will also be of great value. The first document[27] is a compilation of experimental and theoretical data proposed by Los Alamos Scientific Laboratory (LASL) personnel an an adjunct to the Nuclear Safety Guide. It provides not only very useful and valuable information, but it also includes an extensive and rather complete bibliography and gives justification for some of the extrapolations of data given in the Guide.

The other document is a nuclear safety handbook being compiled at the Union Carbide Nuclear Company's Y-12 Plant which provides not only criticality data under a wide range of conditions, including those where the use of more than one control consideration permits increased geometry or masses, but also gives adjunct information that is useful in machine type calculations. In addition, available data are treated in a manner that permits the solution of some problems by reference to simple graphical interpretations.

Standards and Regulations

Activities in the field of government regulations have not resulted in any different or additional basic requirements at the national level from those given in the *Code of Federal Regulations*.[3]

In the United States, activities in the fields of standardization have resulted in the preparation by Subcommittee N6-8 of the American Standards Association (ASA) of a proposed standard entitled "Fissionable Material Outside Reactors" (previously published in *Nuclear Safety*[8]). This standard, which is currently in the process of review, gives no specific technical and operating criteria but refers to the reference documents[18,19] as sources for such data. A few comments are presented on the basic aspects of nuclear safety in production or processing

facilities, including their hazards, together with some idea of the Subcommittee's philosophy on nuclear safety in general. Some possible specific administrative methods are also indicated, although the treatment of this subject is rather general.

At the international level, no action has been taken by the International Standards Organization (ISO). However, the International Atomic Energy Agency (IAEA) has under review and development some proposed criteria that are primarily concerned with shipments. These use the available technical data as a basis. The first draft of this proposal appears to indicate an approach which, in many ways, departs from the criteria and administrative methods that have been generally used in most facilities handling fissionable materials in the United States.

Conclusions

Progress continues to be made in the development of criteria applicable to nuclear safety and in making such information generally available; however, there has been no basic change in the methods that appear to be appropriate for handling the various problems involved. An increasing amount of information is being published in the field; most of this material provides extensions of basic data and frequently includes assumptions which are not always obvious and which give results that must be used with care. This consideration makes it perhaps more important than ever that facility operators review original experimental work and basic nuclear safety criteria prior to design and operation to ensure that the possible hazards in activities involving fissionable materials have been carefully assessed and that there exist no inherent hazards resulting from unrecognized inadequacies in the safety criteria chosen for a particular situation.

(Hugh F. Henry)

*Indemnity for AEC Contractors and Licensees**

Liability insurance and indemnity for atomic energy operations have been discussed from the historical development viewpoint,[28] and the status of federal (AEC) regulations on this subject has been reviewed[29] in previous issues of *Nuclear Safety*. Recently, Hennessey* prepared a description of the evolution of insurance and indemnity practice by the AEC since its establishment in 1947, when it inherited the contracts of the Manhattan Engineer District, which had been the cognizant federal authority over atomic energy work since its inception. A condensation of Hennessey's work is presented here.

Prelude to the Price-Anderson Act

The unknown potential dangers involved in working with atomic energy led contractors of the Manhattan Engineer District to require indemnity by the government over a very broad range of provisions which held the contractors harmless against claims that might arise in connection with their contract activities. When the newly formed AEC took over the program, it concluded that the dangers were real and therefore continued to indemnify the existing contractors and extended similar indemnity to new contractors whose work involved nuclear hazards.[30] It was the Commission's contract policy to limit indemnification by excluding from coverage any occurrence resulting from such causes as gross negligence, willful misconduct, or bad faith of specified contractor representatives, who were normally upper strata company-management officers. The contractors were generally willing to assume the remote risk involved.

A more serious difficulty was the legal limitation that the total funds made available to the AEC by Congress for carrying out its opera-

*The views expressed in this article do not necessarily represent those of the U. S. Atomic Energy Commission or its appropriate office or division.

*This article is a condensation of a chapter prepared by Joseph F. Hennessey for publication in England by Pergamon Press in a volume entitled *Nuclear Liability* of the Pergamon Press *Progress in Nuclear Energy* series. Mr. Hennessey is Counsel, Division of Reactor Development, U. S. Atomic Energy Commission. He previously served as Counsel for AEC's Division of Production and as attorney at the Oak Ridge Operations Office. Prior military service included an assignment with the Manhattan Project as legal adviser to Major General Leslie R. Groves. Mr. Hennessey received the A.B. degree from Holy Cross College in 1931 and the LL.B. degree from Harvard Law School in 1934. He is a member of the Massachusetts bar.

tions and obligations were the maximum that could be committed by the AEC.[31] In entering into indemnity agreements, the Commission had relied upon its general contracting authority either from the Atomic Energy Act[32] or from the First War Powers Act,[33] neither of which provided an exception to that limitation. The Commission therefore customarily included the pertinent limitation in the amount of indemnity available under its contracts. In a few exceptional cases, unlimited indemnity was authorized by the President under his authority conferred by the Act.[34]

Until the enactment of the Price-Anderson bill in 1957, there was an uncomfortable lack of completeness of coverage of AEC prime contractors, and there was much concern among the many subcontractors and suppliers of services and equipment for construction, operation, and maintenance, who remained unprotected against liability. The Commission's indemnity policy was to indemnify only those with whom it had privity of contract, as based on its contractual authority. The prime contractors were naturally unwilling to pledge their own credit to indemnify others. Because of these limitations and the associated incomplete protection of the public, pressure arose for the government to provide effective protection to all by a broader indemnification base.[35,36]

The Price-Anderson Act and the Contractor

The Price-Anderson Act became law[37] Sept. 2, 1957. It established authority for AEC to require that licensees and contractors furnish financial protection as specified by the Commission and for AEC to indemnify them against liability, above the amounts so specified, to a maximum government liability[38] of $500 million. The Act also established the total protection, i.e., the licensee-furnished protection plus the government indemnity, as the maximum legal liability that could be generated by one incident. Similar protection was provided for Commission contractors[39,40] who were exempted from the licensing provisions of the 1954 Act[41] by extending to them the same protection requirements and indemnity as for licensees.

The Price-Anderson Act indemnity authority eliminated three major limitations in previous indemnity provisions:

1. Indemnity now does not exclude incidents resulting from misconduct, bad faith, or negligence of the contractor's officers or employees. The Committee report[42] states that the damage to the public is the same regardless of the cause, and the purpose of the Congress was to protect the public, leaving punishment of malefactors to other channels.

2. The limitation of coverage to the amount of funds appropriated was removed,[43] "without regard to Sec. 3679 of the Revised Statutes, as amended," thus firmly pledging the credit of the United States to the indemnity undertaking, limited now only to the private protection plus $500 million government indemnification.

3. The new (Price-Anderson) act protects the public not only from incidents caused by the prime contractor but by his subcontractors and suppliers,[39,44] and indeed by any person[45] who might become liable, regardless of contractual relation. Only acts of war are excluded.[44] This significantly increased the willingness of subcontractors and suppliers to do business with licensees and contractors.

To emphasize the purpose of the Act, the Joint Committee expressed the desire that the Commission adopt a policy of extending the discretionary indemnity provisions to contractors and their subcontractors, as well as to licensees and their subcontractors.[40]

Commission Implementation of Price-Anderson Act Authority

The Commission acted promptly to demonstrate its intention to implement the provisions of the new Act. In December 1957 the Commission announced that it was prepared to enter indemnity agreements under the Act with any of its affected prime contractors, and its contract administrators were instructed to offer Price-Anderson indemnity to all eligible contractors, old and new. From the resulting discussions with contractors, there evolved the basic form of indemnity agreement now offered to Commission contractors.[46]

The agreement generally reflects three basic concepts underlying Commission policy in the application of Price-Anderson Act indemnity authority: (1) that the full protection afforded by the Act should be extended to the eligible contractors and, through them, to the public; (2) that, to the extent feasible, any nuclear incident which may occur should be covered by one, and only one, Price-Anderson Act indemnity agreement, which is to say that the

agreements should be such that only one contractor or licensee could be held liable for any one incident; and (3) that the indemnification of contractors should provide protection that parallels, but does not overlap, the protection under license indemnity.

Extent of Coverage. Not all contractors of the Commission are eligible for Price-Anderson Act agreements. Section 170(d) of the Act[39] provides, in part:

> In addition to any other authority the Commission may have, the Commission is authorized until August 1, 1967, to enter into agreements of indemnification with its contractors for the construction or operation of production or utilization facilities or other activities under contracts for the benefit of the United States involving activities under the risk of public liability for a substantial nuclear incident.

"Substantial nuclear incident" is not specifically defined in the Act. Many contractors are engaged in activities which resist definition as operation of "production facilities" or "utilization facilities," as for example the refining of "source material"[47] (e.g., uranium ore), or the fabrication of reactor fuel elements of "special nuclear material"[48] (e.g., enriched uranium), or the utilization of "by-product materials"[49] (e.g., radioactive isotopes). Each of these activities involves some risk of occurrence of a nuclear incident. The Commission has recognized the need for an objective definition of substantial nuclear incident and therefore for technical standards by which to define it in order to provide uniformity of application of its statutory authority.[50] It is continuing studies directed toward defining such standards.

Meanwhile, the *modus operandi* of the Commission has been to offer full indemnity if its contracting representative concludes that $60 million or more damage could conceivably result from an incident arising in the course of the contractor's operations and no indemnity if he concludes that the damage could not exceed $1 million. Most cases fall within one or the other of these two categories. For the unusual cases of estimates of between $1 million and $60 million maximum conceivable damage, the Commission's General Manager has retained the authority to determine the extent of risk and to require that the amount of financial protection so determined be furnished by the contractor.

The indemnity agreement protects the contractor for any nuclear incident occurring at a contract location, whatever the cause of the incident.[46] "Contract location" includes any Commission site or contractor site at which the contractor is active under the contract terms. In addition, the contractor is protected for contract activities at other locations by his employees or by anyone for whose activities he is legally responsible. Nuclear incidents arising from transportation of nuclear materials to or from a contract location are covered. The contractor is also protected against products liability, defined to include equipment, materials, facilities, and design or other technical data. As a further safeguard the agreement provides that the Commission's obligations shall not be affected in the event of the contractor's breach of contract, death, disability, or termination of existence, or by the termination or expiration of the contract.

All nuclear hazards involved under a contract are covered to the extent of the maximum hazard, regardless of the estimated degree of the other hazards. That is, having found the maximum hazard to be sufficient for coverage, the Commission covers it and all other hazards of whatever degree connected with the contract with the same indemnity provisions.

Since contractors objected to the situation that indemnity was limited, by definition, to incidents in the United States,[51] whereas the old contracts covered potential liability for foreign incidents, the Commission soon decided it would be necessary to permit its prime contractors to retain the broader rights of the existing indemnity agreements while extending to them the full benefits of Price-Anderson Act indemnity. Although it is the Commission's policy to include only Price-Anderson Act indemnity in new contracts, exceptions are sometimes granted when it is determined that a contractor will be exposed to risk of liability outside the United States. The Commission staff has under consideration a possible expansion of its policy to cover those of its contractors who, although having no substantial risk in performing their own contract activities, may deliver a product to an unindemnified user[52] (e.g., the Department of Defense) whose use of the product might expose the furnishing contractor to risk of liability from a substantial nuclear incident.

These considerations demonstrate the Commission's policy of attempting to extend full

Price-Anderson Act protection to the public through agreements with its licensees and contractors. Evidence of the fruition of that policy is found in the fact that every major Commission installation involving the risk of a substantial nuclear incident is now covered by a Price-Anderson Act indemnity agreement.

Singleness of Coverage. A basic principle of Commission policy is that any nuclear incident should be indemnified under only one agreement. The Price-Anderson Act makes this possible by defining the "person indemnified"[53] as not only the person with whom the agreement is executed but also "any other person who may be liable for public liability" because of activities in connection with the activity indemnified. For example, the Commission normally will make an indemnity agreement only with the operator of a reactor, but in such terms that all subcontractors, suppliers, and any other persons who might be found liable for a substantial nuclear incident connected with the reactor would be covered by the same agreement. This method of approach was suggested by the insurance companies to prevent pyramiding of coverage as a result of having each of the designers, owners, contractors, and others interested take out separate policies.[54] This principle has been well received by industry. For example, in the case of the Commission's varied production and research activities at Oak Ridge, Tenn., with literally thousands of subcontractors and suppliers, all are content to rely upon the indemnity agreement contained in a single contract between the Commission and the operator of the installation. Each may be assured of the protection of Price-Anderson Act indemnity by obtaining a representation from the prime contractor or the Commission that he is covered as a "person indemnified" under the Act. In this way the difficult legal problems of having the same incident involved in numerous simultaneous agreements are avoided.

To prevent a lack of coverage of persons (architects, engineers, constructors, suppliers, etc.) furnishing supplies or services before an installation is covered by Price-Anderson Act indemnity, the Commission gives formal contractual assurance to each contractor and other persons during the construction period that it will enter into a Price-Anderson Act indemnity agreement covering the operation of the facility or, lacking such an agreement, will enter into a direct agreement. To avoid a time gap in coverage, the contract includes assurance that the individual indemnity agreement will be retroactive to the time when the risk of liability for a nuclear incident arises.

It is possible that one indemnified contractor doing work for another might be indemnified, not under his own contract, but under the other's contract. For example, a fuel-element manufacturer supplying fuel elements to a reactor operator might be indemnified under the operator's contract against an incident caused by a faulty fuel element.

Contractor Indemnity Parallel to Licensee Indemnity. The third aim of the Commission has been to provide parallel, but not overlapping, indemnity to contractors, equivalent to that offered to licensees. This aim is in response to an expression of desire by the Joint Committee[40] that the two cases be treated consistently, and it indicates the Commission's endorsement of the Congressional intent to give maximum protection to the public.

The contractor is afforded all the advantages of the licensee indemnity authorization. It turns out, in fact, that his coverage is even broader than that available to the licensee. He is covered with respect to "contractual activity," whereas the licensee is covered with respect to "licensed activity," which is normally localized at a particular site. Contractual activity is more transient in nature, following the contractor's employees as they move from site to site under the contract, and is more enduring in the potential effects of its products.

Contractor indemnity does not overlap licensee indemnity. The standard agreement[46] specifically excludes Price-Anderson Act indemnity coverage for any risk already covered under an active agreement with a covered licensee.

Unindemnified Subcontract Activities

The standard indemnity agreement[46] protects the subcontractor only with respect to work performed at a "contract location,"[55] which is defined to include only facilities of the Commission and the prime contractor. The subcontractor is indemnified with respect to his products delivered to the prime contractor, but not for his activities carried out elsewhere than at a contract location. Since the Commission's indemnity authority extends to "public

liability arising out of or in connection with the contractual activity,"[39] statutory interpretation is involved. It is the Commission's position that, if work is performed under license, any Price-Anderson Act indemnification that is extended must be that available under the licensing program. The alternative would be to permit all licensed handlers of nuclear materials to become beneficiaries under the contractor indemnity program and thus negate the express intent of Congress that licensees should only be indemnified under the conditions established in the Price-Anderson Act for indemnification of licensees.

Contractor Insurance

The Commission is authorized to require its contractors to furnish financial protection against public liability for nuclear incidents.[39] However, since the Commission determined that the probable cost of insurance which the government would have to bear (in excess of $10 million a year) would not be justifiable in the light of its estimate of the risk, the standard indemnity agreement contains no requirement for the maintenance of financial protection by the contractor. The Commission does, however, reserve the right to require such protection. The Price-Anderson Act directs that the Commission use the services of private insurance organizations to the maximum extent available[56] in administering the Act. The industry has made its claims-adjusting services available to the Commission, and the Commission is negotiating with the industry. One agreement for services has been executed, and others are being negotiated.

Joint Projects

The provision that the Commission may indemnify its contractors in "contracts and projects financed in whole or in part by the Commission"[39] was intended to make available indemnity protection for contracts and projects the Commission might enter into jointly with other government agencies. In particular, Congress was cognizant of development activities for nuclear-propelled submarines and aircraft being carried on as joint projects of the Commission and branches of the armed services, in which research and development work was done under a Commission contract and the vehicle itself was constructed under contract with the military agency.[40]

In making indemnity agreements under this authority, one Commission technique has been to include in the indemnity agreement with its contractor a statement of understanding to the effect that a nuclear incident connected with the contractor's work for the military agency will be considered to have occurred in connection with the contractor's activities under its contract with the Commission. It is expected that persons who may become liable in connection with these contracts will be protected even though an incident occurs subsequent to completion of the Commission contract and delivery of the product to the military agency concerned.[50]

Since a nuclear reactor acquired by military agencies for military purposes is exempted from licensing requirements by the Atomic Energy Act, it is exempt from Price-Anderson Act requirements of financial protection and indemnification of licensed reactors. The Commission's indemnification of its contractors' participation in a joint project extends some measure of protection in an area which otherwise lacks protection of the type afforded by Price-Anderson Act indemnity.

The Present Situation

It may be concluded with considerable assurance that the Commission's policies in exercising the authority of the Price-Anderson Act have provided a completely effective shield for its contractors against the threat of public liability arising out of a nuclear incident within the United States. The Commission has entered into Price-Anderson Act indemnity agreements with 43 of its prime contractors. These agreements cover all the AEC installations where there exists the hazard of a substantial nuclear incident. One of these agreements* covers any nuclear incident that may arise in the course of transportation of materials for the Commission over the main lines of railroad transportation in the United States. By means of these agree-

*Agreement with the Chairman, Traffic Executive Association–Eastern Railroads, the Chairman and Counsel, Executive Committee–Western Traffic Association, and the Chairman, Southern Freight Association, acting on behalf of the railroad carriers who are parties to Consolidated Freight Classification No. 21 and Railway Express Agency, Inc.

ments, the public has complete protection against the consequences of a major nuclear incident occurring at any of those facilities where the AEC carries on its varied production and research activities. Protection is provided, too, for all contractors, subcontractors, and suppliers who in any way contribute to the construction or operation of those facilities. Thus, in the area of the Commission's contract activities, the dual objectives of the Price-Anderson Act appear to have been effectively achieved. (L. A. Mann)

Physical Aspects of Radiobiological Damage

As the development of nuclear power depends on nuclear physics and the understanding of biology depends on studies of cells, their nuclei, chromosomes, and DNA (desoxyribonucleic acid), so does achievement of nuclear safety through health-physics and management procedures ultimately depend on basic radiobiology. A complex chain of events precedes most gross biological manifestations of ionizing radiation treatment. In scope, size, and time, this sequence ranges from direct, immediate physical interactions of radiation with matter on an atomic or subatomic scale in perhaps 10^{-17} sec, through indirect physical and chemical effects, biochemical effects at molecular and macromolecular levels, and finally to biological effects on even the most highly developed organisms, where expression of delayed somatic and genetic effects may require decades. The intricacies and extent of this sequence make radiobiology a broad field involving techniques and concepts of conventionally diverse disciplines. This article surveys some of the physical aspects of the radiobiology of ionizing radiations.

Dosimetry

The need for both physical criteria and units to characterize radiation exposure has given impetus to the development of radiation dosimetry. Since radiation produces biological effects only by transferring energy to the irradiated system, a direct measure of the amount of any radiation exposure is the absorbed dose or amount of energy dissipated by the radiation per unit mass of irradiated material. Internationally, the currently accepted unit for absorbed dose is the rad, a quantity having a magnitude[57,58] of 100 ergs/g. The absorbed-dose concept impartially includes all forms of energy transfer to the irradiated atoms and molecules (ionization, excitation, translation, rotation, and vibration). Apart from different spatial distributions, these interactions might be expected to be present in varying proportions, depending on the radiation and the irradiated system, and thus might cause biological responses that differed in kind and frequency. This hypothesis accounts for marked radiobiological differences between X rays and visible or ultraviolet light, which cause no ionization. However, the radiobiolcgical differences between ionizing radiations are not generally attributable to the different proportions of energy-transfer mechanisms, in view of the fact that (1) the interactions of ionizing radiation and charged particles are nonspecific statistical interactions and (2) comparable experimental values of the ratio, W, of energy per expended ion pair produced in different gases by charged particles are of vastly different energies. Rather, differences in radiobiological sensitivities of ionizing radiations are attributed to variations of the biological system or spatial distribution of energy deposition.

Next we briefly consider operational problems of dosimetry. Direct measurement of absorbed dose, preferably by calorimetry, is inherently impractical because of what has been called the central problem of radiobiology, namely, the great sensitivity of biological systems to radiant energy. For instance, the energy deposited by radiation lethal to man is thermodynamically insufficient to raise his temperature 0.01°F. Hence one uses indirect but more sensitive measurements, such as ionization of gases, darkening of photographic emulsions, fluorescence of phosphate glass, nuclear activation of selected elements, and colorimetric changes of certain chemicals. Refinements of devices and their absolute calibration are described in textbooks and extensive literature.[59,60] A common concern in physical dosimetry and fundamental radiobiology is a working knowledge of interaction mechanisms between radiation and matter.

Interactions of Radiation with Matter

Electromagnetic or photon radiations of atomic origin (X rays) or nuclear origin (gamma

rays) interact by: (1) the photoelectric effect, whereby a photon imparts all its energy to one electron;* (2) the Compton effect, whereby the incident energy is divided between an ejected electron and an emergent photon; and (3) pair production, whereby the photon is converted to an electron and a positron with kinetic energy and mass totaling the initial energy. As the photon energy increases, the predominant effect shifts from the photoelectric effect to the Compton effect at about 30 kev and to pair production at about 3 Mev. Instead of chemical bonding, the electron concentration and, for photoelectric effect and pair production, the atomic number are the significant target parameters of tissue dosimetry. Each process directly produces charged particles of appreciable energy, the primary effect of photon irradiation.

All neutrons interact primarily with atomic nuclei by such reactions as (n,p), (n,d), (n,α), (n,n), and (n,γ). Charged particles result even in elastic or inelastic neutron scattering because, except for low-energy neutrons, the velocity of the recoil causes stripping of at least one electron. Emission of characteristic gamma rays from nuclei raised to excited states indirectly produces more charged particles. The probabilities of particular reactions depend principally on the target nuclei and incident neutron energy. For neutron energies from 1 to 10 Mev in typical biological tissues, elastic scattering by hydrogen, because of preponderance of hydrogen nuclei and relatively large energy transfer to these nuclei, contributes more than 80 per cent of the absorbed dose. For thermal neutrons or neutrons above 10 Mev, other reactions become important.[61–63]

Thus for energetic photons and all neutrons the significant direct interaction is production of energetic, charged particles. Almost all biological radiation damage is thought to result from charged particles traversing tissues. In passing through matter, each particle dissipates its energy by many electrostatic interactions with orbital electrons and almost never with atomic nuclei (except at extremes of energy). Dissipation is about equally divided between ionization and excitation, although at highly relativistic velocities the bremsstrahlung, or "stopping radiation," becomes important. Formulation for the linear rate of energy loss of charged particles, expressed as $(-dE/dx)$ in physics and as linear energy transfer (LET) in radiobiology, was made almost three decades ago.[64] For strongly relativistic particles, polarization effects in dense media decrease the rate of energy loss to 4 per cent less than in gases.[65] Here we leave fairly well-understood, direct physical interactions and consider less well-understood, indirect actions, where competing hypotheses may explain events which are probable but which are not directly verified by experiment.

Models of Radiobiological Action

Further explanation of the radiobiological sequence is largely in terms of partially proven concepts of the significant action. Because of the scope of the systems and end points studied, no single concise model should be expected to explain all radiobiology. Generally the models imply direct or indirect action. In the direct action, one first assumes that energy is absorbed directly by a critical molecule of the irradiated cell and then considers the results of such absorption on the molecule and its reactions with its environment. In Dessauer's original presentation,[66] the energy transfer was considered as thermal. He used the term "point heat," but most modern workers consider the significant energy transfers to be ionization and excitation. The shock of ionization in a protein structure was discussed by Platzman and Franck,[67] who found that the resulting polarization was likely to degrade profoundly the molecular organization.[67] Indirect action, wherein energy absorbed at a distance is transferred to a critical molecule via chemical intermediaries, often free radicals, might also be considered as cellular poisoning. Since water is almost omnipresent in biological systems, the radiation chemistry of water, which Hochanadel[68] has recently reviewed, with all its detailed reaction kinetics for the production and decay of ions and radicals, becomes important for radiobiology. Ebert[69] observed that radiobiology tends to lie between the extreme

*Ultraviolet and visible photons dissipate their energies by single molecular collisions. Because of quantized states of electrons, specific absorptions occur only at particular energies, thus making possible many analyses of chemical structure or site of biological action. In a few alkali metals with low work function, these radiations produce photoelectrons.

radiochemical examples of direct action on pure substances and indirect action on dilute solutions. Hence the hypotheses of combined action, such as those of Zirkle and Tobias[70] and Wijsman,[71] are natural developments despite difficulties in experimental application.

Radiobiological Examples

Whether one thinks of direct or indirect action, some variant of the target theory, for which perhaps Lea[72] receives the most credit, is commonly invoked, largely because it offers a qualitative reason for the high sensitivity of biological systems to ionizing radiation. One assumes that observed biological effects stem from ionization and thereby damage to, or inactivation of, a strictly limited number of biologically sensitive, highly specific sites or targets. As Fano[73] remarked, "... macroscopic biological phenomena are steered to a great extent by submicroscopic structures ..."

An elementary application of target theory, assuming direct action of the radiation, is size determination of radiobiological targets. Assuming that each hit inactivates an enzyme molecule, then the rate of inactivation with a measure of dose, dN/dE, and surviving fractions, N/N_0, may be expected to be

$$dN/dE = -NS \tag{1}$$

$$\ln N/N_0 = -SE \tag{2}$$

If E is the flux across randomly distributed targets, S represents the average cross-sectional area. Similarly, if E represents primary ionizations per unit volume, S is the volume of the target. Pollard and associates[74] have long exploited this technique and *inter alia* found the target size for several enzymes to be just the entire molecule. Conversely, assuming this principle, one may calculate approximate molecular weights from target sizes.

Using curves of biological effect versus dose, one can describe the radiobiological sensitivity and sometimes partially describe the initiating action. Thus the linearity of the logarithm of the surviving fraction of microorganisms versus dose indicates that only one hit or inactivation is fatal to the cell; however, under certain circumstances when the survival curve is convex, one can conclude that the number of interactions required to kill the cell is that indicated by the intercept at zero dose. Atwood and Norman[75] have thoroughly studied the interpretation of survival curves.

Continuous studies of chromosomal aberrations produced by ionizing radiations illustrate general trends. One microscopically observes fragments and rearrangements of chromosome strands produced by irradiation and notes that their frequencies per unit dose depend on both dose rate and LET. Qualitatively, target theory predicts such results if one assumes that several ionizations are required to produce a single primary chromosome break and that most primary breaks naturally rejoin so as to become unobservable. For equal doses, the probability of the required concentration of ionizations occurring at or near a site of potential chromosome breakage would be expected to increase with LET until the LET becomes so high that, whatever part of the track crosses the chromosome thread, much of the ionization is wasted. Conger et al.[76] presented such data for chromosomal aberrations in the flowering plant *tradescantia*. Quantitative, mechanistic interpretation depends on the choice of LET characterization, "track average," "energy average" (Boag, 1954), or Rossi's Y concept (1959).[77,78]

If multiple hits in a limited time must precede observable aberrations, an increase in aberration frequency with dose rate over some range would be expected, at least for low LET irradiations. For high LET radiations, where the localized multiple ionization from a single particle may cause an aberration, less dose-rate effect is expected. Such considerations, supported by experiments, plus the generalization that availability of oxygen accentuates many radiobiological effects,[79] lead to the more complex view expressed by Alper,[80] namely, that radiobiological sensitivity may be greatest for doses delivered faster than cellular repair processes can occur and slower than the time needed for oxygen to diffuse to or within a cell, thus avoiding local anoxia during irradiation. Bacq and Alexander[81] cite unstable products and radical-radical interactions as possible further cause of dose-rate effects. Wolff and Luippold[82] emphasized the role of the "rejoining system" in the yield of observed aberrations.

Conclusions

Radiobiology, proceeding both from the foundation of physical comprehension of immediate

interactions, such as ionization, and from the biological foundation of microscopically demonstrable events, such as chromosome fragments, is increasingly concerned with intermediate mechanisms thought to be of complex physical-chemical nature. Concepts that involve a balance of factors simultaneously acting, rather than the simple approach of direct action on a geometrically describable target, are currently prominent. However, the simple target theory and direct action theories still have places in radiobiology at least pedagogically comparable to that of the Bohr theory of atomic structure.

Generalizations of LET and dose-rate phenomena to other biological systems often apply, but there are exceptions. One of the most striking is that purely chemical systems show a decrease of sensitivity with increasing LET.

The breadth of the general subject, limited space, and the author's attempt to present a coherent picture made this presentation sketchy. Many details and qualifying statements have been omitted, all points of view were not presented, and some significant phenomena and studies were not discussed. Pertinent literature considers biological variability in both cells and organisms and between individuals, which forces radiobiology to rely heavily on statistical analysis; emphasizes competing, especially repair, biological mechanisms; discusses synergistic and antagonistic effects of combined treatments of various radiations and other chemical or physical agents; attempts to correlate electron-spin measurements with radiobiological effects; and analyzes diffusion and migration of ions and radicals. All these and other matters enter the physics of radiobiology. (M. L. Randolph)

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Section II

ACCIDENT ANALYSIS

Integrity of UO_2 Fuel Elements

Three barriers are provided to prevent the dispersal of radioactive materials from power reactors—the fuel elements, the reactor system, and the containment vessel. The first line of defense against release of radioactivity to the environment is provided by the fuel elements. Therefore, in safety considerations, it is important to understand the mechanisms by which fuel elements can fail and the results of such failures. Previous articles in *Nuclear Safety*[1,2] have reviewed the problem of the escape of fission products from irradiated UO_2, and one article[3] dealt with fuel-element failures, including some in UO_2 fuel elements. The purpose here is to review some aspects of the failure mechanisms which have been observed or postulated for bulk UO_2 fuel elements.

World acceptance of bulk UO_2 as a reactor fuel is due to several desirable characteristics, including (1) reproducibility of high-quality sintered fuel bodies, (2) compatibility with a variety of cladding materials and coolants, (3) high melting point, and (4) stability at relatively high burnups. Serious investigations of UO_2 as fuel material began with the advent of the Shippingport Pressurized-Water Reactor (PWR) development program. Since then, significant advances in UO_2 technology have been made, primarily as a result of empirical irradiation testing of fuel elements for specific reactor applications. In particular, the bulk of available information is concerned with sintered UO_2 in cylindrical rods for use in water-cooled reactors. Consequently a large portion of this review is concerned with cylindrical fuel elements containing sintered UO_2; however, some information on flat-plate elements and nonsintered UO_2 is presented.

Effects of Cladding Defects

Present power-reactor cores contain several miles of fuel elements, and the expectation that this amount of material could be fabricated defect-free would not be realistic; even if no defects existed when the reactor core was first assembled, small defects could develop sometime during the core lifetime. Therefore a fuel element that is subject to gross distortion or rupture as a result of a small defect would not be desirable. In-pile tests of fuel elements with known defects have been made and have provided much information on the effects of defective UO_2 fuel elements in water systems, including "waterlogging" and, with Zircaloy-clad elements, hydriding of the cladding.

Extensive distortion and rupture of a fuel element containing a small defect can occur in water systems by a mechanism known as waterlogging. A waterlogging failure results when water enters the fuel element during reactor shutdown, and upon subsequent startup the water flashes to steam and causes excessive pressure within the element. Eichenberg and coworkers[4] have listed three conditions that could lead to a failure of the cladding of a defective element: (1) a small defect could severely restrict the escape of steam, (2) the original hole could be blocked by a piece of UO_2 that would restrict the escape of steam, and (3) water in open pores of the UO_2 could cause violent fragmentation of the fuel during heatup.

Concern over the possibility of waterlogging failures in the first PWR core resulted in a series of in-pile defective-element tests.[4] Twenty-four elements containing 5-mil holes were irradiated. Several of the rods showed diametral expansions of up to 12 mils; however, only one specimen actually failed. This specimen, which apparently failed because of violent

fracture of the UO_2 (condition 3 above), contained 80 per cent dense UO_2. As a result of these tests, it was concluded that waterlogging would not be a significant problem for the PWR core 1 fuel elements if UO_2 of approximately 93 per cent of the theoretical density was utilized.

A Chalk River[5] report states that, in more than 30 loop tests on rod type UO_2 specimens conducted up to May 1959, only two waterlogging failures occurred: (1) the PWR failure discussed above and (2) the failure of a nonstoichiometric UO_2 specimen clad in stainless steel. This latter specimen was not intentionally defective, but it contained a small crack that penetrated the wall and was enlarged during irradiation.

Results of defect tests on compartmented-plate fuel elements containing sintered UO_2 have also been reported.[6] Four compartmented fuel plates (each plate composed of several compartments) were made defective by drilling a 40-mil hole through the cladding. None of these defective compartments failed; however, a compartment adjacent to one of the defective compartments failed because of waterlogging after about four days of irradiation. Investigations later revealed that the compartment which failed and the defective compartment were interconnected by pores in the rib separating the two compartments. The conclusion was that the 40-mil holes were sufficiently large to allow steam to escape without significant pressure buildup, whereas the pores leading to the failed compartment severely restricted steam expulsion. It seems reasonable that plate type elements would be less resistant to waterlogging failures than rod type elements. This apparently was one of the factors that influenced the Canadian choice of cylindrical-rod fuel elements for CANDU.[7]

Waterlogging failures of nonsintered UO_2 fuel (as fabricated by swaging or vibratory compacting) would appear to be particularly undesirable if gross release of particulate matter were to occur. Two swaged UO_2 fuel rods containing 5-mil defects in the Zircaloy-2 cladding were irradiated for approximately 2 hr, and no dimensional change or release of particulate matter was observed.[8]

Thus a large number of irradiation tests of UO_2 fuel specimens has been performed, and the available information indicates that only four specimens have failed as a result of waterlogging. However, two of those failures occurred as a result of *unintentional* defects. This appears to demonstrate that it is difficult to perform meaningful waterlogging tests. A defect small enough to produce waterlogging is likely to be a "natural" defect that might be overlooked during inspection. It appears that the conditions which tend to make waterlogging important are small defects, large void volume inside the fuel element, and fast reactor power transients. The reviewer knows of no investigations that have attempted to establish allowable combinations of these conditions, and such an investigation would seem to be warranted.

Defective UO_2 fuel specimens clad with Zircaloy-2 and operated in water reactor systems are particularly subject to hydrogen absorption in the cladding. Localized, high concentrations of hydrogen at the external surface of the cladding have caused embrittlement and cracking.[4,9,10] This is probably one of the most important mechanisms for failure of Zircaloy-2-clad fuel elements. The hydriding phenomenon in defective elements has been postulated to occur in the following manner:[11,12]

1. During reactor operation a steam atmosphere would exist in the interior of a defective element.
2. The steam atmosphere would slowly be replaced by hydrogen through the corrosion reaction of steam with the Zircaloy-2 cladding.
3. In areas of high hydrogen concentration, presumably in pockets at the UO_2-to-cladding interface, gross amounts of hydrogen would be absorbed by the Zircaloy-2.
4. Since a thermal gradient would exist in the cladding during reactor operation, the hydrogen would diffuse down this gradient and form a brittle $ZrH_{1.5}$ layer at the outside surface, thus reducing the strength of the cladding.
5. Subsequent pressure surges in the element would then produce failure.

The results of several tests[10] have indicated that the necessary conditions for hydriding type failures are a cladding defect, small clearance between UO_2 and cladding, and high heat flux. A summary of these tests is presented in Fig. II-1.

Tests of Zircaloy-4 (which contains less nickel than Zircaloy-2) have shown that it absorbs much less hydrogen than Zircaloy-2. The use of Zircaloy-4 as fuel cladding material would appear to alleviate the hydriding problem.

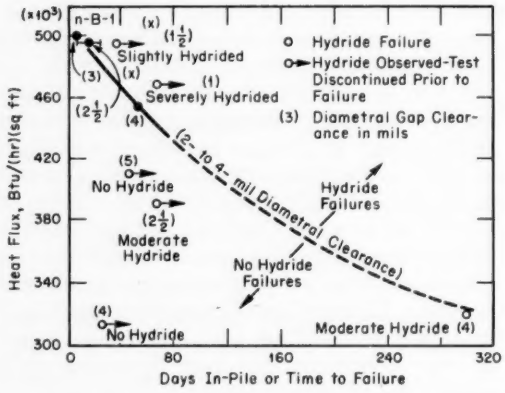


Fig. II-1 Hydriding of Zircaloy-2 in defective PWR core 1 blanket fuel-rod samples in in-pile high-temperature water loop. No hydriding or failures were observed in rods with 8-mil diametral clearance; these rods are indicated by (x).[10]

Burnup Limitations

The dimensional stability of UO_2 at relatively high burnups is an important asset of this fuel. Investigations directed toward PWR fuel development revealed that there were no appreciable dimensional changes in UO_2 fuel elements[4] after burnups as high as 25,000 Mwd per ton of UO_2. Until recently, no upper limit on UO_2 burnup had been established, although work on fully enriched UO_2 dispersed in metal matrices gave evidence of irradiation-induced swelling of the UO_2 particles.[5] Recently, irradiation testing of plate type sintered UO_2 fuel elements has given evidence that a definite burnup limit may exist for bulk UO_2.[6] Prototype PWR core 2 fuel elements containing UO_2 fuel platelets sintered to 96 to 98 per cent of the theoretical density were irradiated at Chalk River to exposure levels as high as 60,000 Mwd per ton of UO_2. Results from the in-pile tests have shown that UO_2 swells about 5 per cent in volume at burnups exceeding 44,800 Mwd per ton of UO_2. For UO_2, which had begun to swell and then was exposed to water as a result of a cladding defect, the volume increased about 25 per cent. In addition, fission-gas release from the fuel at 60,000 Mwd per ton of UO_2 was about 10 per cent; this is to be compared with a gas release of 1 per cent for the same fuel at exposures up to 22,000 Mwd/ton. Corresponding to the dimensional changes, there appeared to be a breakdown of the UO_2 lattice structure, as revealed by X-ray and metallographic examinations.

It was observed that the phenomena associated with high burnup were not linear with exposure but occurred at a threshold exposure. The results of the observed dimensional changes are

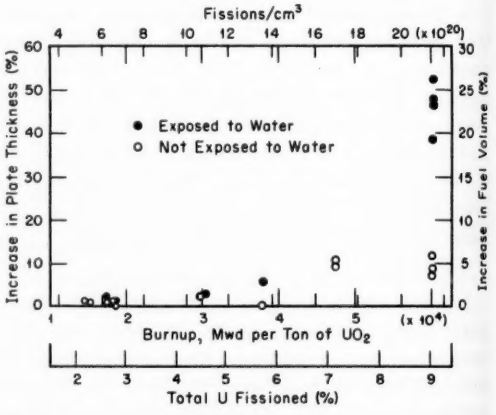

Fissions/cm³
4 6 8 10 12 14 16 18 20 (x10²⁰)
Increase in Plate Thickness (%)
Increase in Fuel Volume (%)
Exposed to Water
Not Exposed to Water
Burnup, Mwd per Ton of UO₂
(x 10⁴)
Total U Fissioned (%)

Fig. II-2 Average thickness changes in irradiated UO_2 fuel plates in the X-3 loop test.[6]

presented in Fig. II-2. As shown in Fig. II-2, the threshold exposure for extensive UO_2 growth is between 40,000 and 50,000 Mwd per ton of UO_2.

In another Bettis experiment,[6] two fuel elements were irradiated at Chalk River starting in April 1958, with interim examinations at approximately six-month intervals. The dimensional data from these experiments are shown in Figs. II-3 and II-4, which give the percentage increase in fuel thickness as a function of burnup. Fuel element I-21 (Fig. II-3) had 21-mil cladding and plate compartments ½ in. wide and 3 in. long, whereas fuel element J-23 (Fig. II-4) had ¼- by 3-in. compartments with 24-mil cladding. Element I-21 underwent a significant increase in swelling at a burnup of about 20×10^{20} fissions/cm^3 (~56,000 Mwd per ton of UO_2); whereas the breakaway point for fuel element J-23 was nearly 30×10^{20} fissions/cm^3. The difference in dimensional stability of the two elements was attributed to the difference in restraint exerted by the cladding. The implication is that, with the proper restraint, UO_2 fuel elements could be operated at burnups exceeding the "threshold." However, as stated in reference 6, "... the degree of restraint necessary to permit exposure of bulk UO_2 fuel ele-

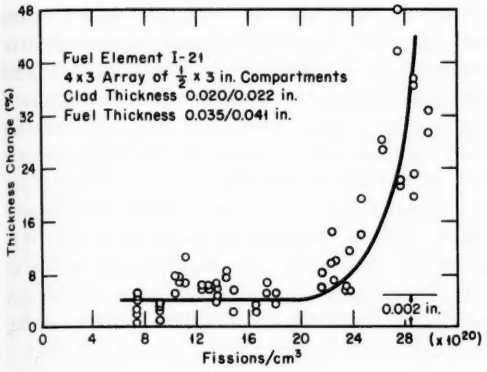


Fig. II-3 Increase in thickness of UO_2 fuel filler as a function of burnup.[6]

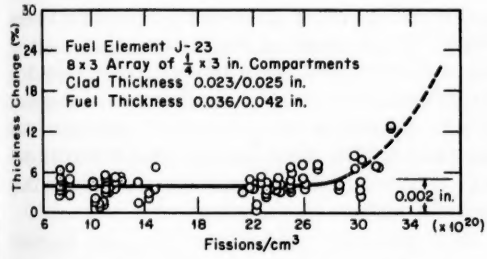


Fig. II-4 Increase in thickness of UO_2 fuel filler as a function of burnup.[6]

ments above the levels which induced the structural changes noted here is, as yet, unknown."

Ratcheting

In almost all in-pile tests of fuel elements containing sintered UO_2, it has been observed that the UO_2 fractures as a result of thermal stresses. Additionally, there is usually a void space between the oxide and the cladding, although this void space tends to disappear when the element is heated because of the differential expansion between the UO_2 and cladding. Based on the characteristics of UO_2 fuel elements described above, a mechanism known as "thermal ratcheting," which could result in cladding deformation, has been postulated to occur in the following manner:[4]

1. As the reactor is brought to power, the UO_2 could fracture as the result of thermal stresses, and the clearance between the fuel and cladding would close up.
2. Upon reactor shutdown, the void spaces inside the fuel element would be reestablished.
3. Small pieces of the fractured UO_2 could relocate into the void spaces.
4. Further cycling of the fuel element could cause deformation and eventual rupture of the cladding.

The PWR fuel-development program included several tests, both in-pile and out-of-pile, to determine the effects of thermal cycling.[4] Out-of-pile tests indicated that the UO_2 did not continue to fracture after the first few cycles, and no dimensional changes in the test specimens were observed. Several in-pile tests were then performed which confirmed the out-of-pile results; no significant dimensional changes were observed after 7000 cycles at a maximum heat flux near PWR design conditions, and thermal fracturing did not appear to progress beyond the initial fractures. Irradiation experience with sintered UO_2 fuel elements at Chalk River[12] has also indicated no fuel-element instability as a result of thermal ratcheting. In addition, Hanford reports[8] that ratcheting has not been observed in swaged UO_2 fuel elements. The information presently available therefore indicates that thermal ratcheting is not a serious problem.

Temperature Effects

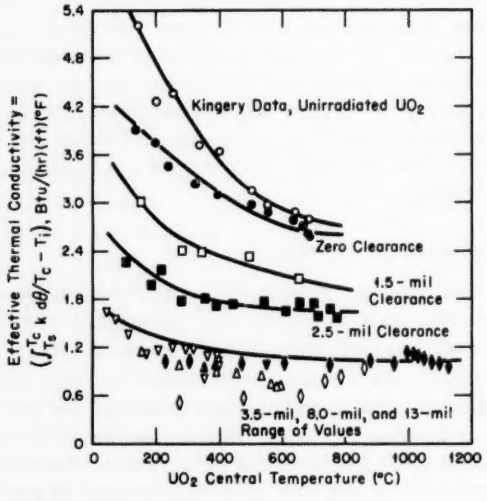
Many of the limitations on the performance of UO_2 fuel elements are associated either directly or indirectly with the fuel temperature. Fission-gas release, differential thermal expansion between the fuel and cladding, and central melting are some of the phenomena which could limit fuel-element performance. Considerable difficulty has been experienced in predicting fuel-element temperatures based on out-of-pile measurements of UO_2 properties. The difficulties have been due, in part, to unknown effects of UO_2 cracking, fuel-to-cladding interface conductance, burnup, and the filler gas. Several out-of-pile thermal-conductivity measurements[13–16] have been made on UO_2. A recent Bettis report[17] gives the results of in-pile measurements of the "effective" thermal conductivity of cold-pressed and sintered UO_2 with several initial clearances between fuel and cladding and various filler gases; these data are shown in Fig. II-5, along with the Kingery[15,16] out-of-pile data. The effective conductivity includes the thermal resistance at the fuel-to-cladding interface and is defined as

$$k' = \frac{\int_{T_s}^{T_c} k\, d\theta}{T_c - T_i}$$

where

k' = effective conductivity
k = thermal conductivity of UO_2 (a function of temperature)
T_C = UO_2 central temperature
T_i = temperature of inside surface of cladding
T_S = surface temperature of UO_2

Some conclusions that were drawn as a result of the in-pile investigations are (1) the initial diametral clearance between fuel and cladding is very important to thermal performance; (2) the conductivity of UO_2 is unaffected by burnups up to 4×10^{19} fissions/cm^3, at least for fuel temperatures above 500°C; and (3) the effective conductivity is unaffected by the filler gas. This last conclusion is rather surprising, since it has usually been assumed that the thermal performance deteriorates during fuel-element life as a result of dilution of the original filler gas (usually helium) by low-conductivity fission gases. If the conclusion is valid at all, it is probably valid only for high-density sintered UO_2. Investigations at Hanford[8] have shown that the filler gas has an important effect on the thermal performance of nonsintered UO_2 fuel.



○ Kingery Unirradiated Data

● WAPD 22-11: Zero Diametral Clearance, 1 Kr + 3 Xe Atmosphere, Production UO_2

□ WAPD 22-4: 1.5-mil Diametral Clearance, Helium Atmosphere, Production UO_2

■ WAPD 22-14: 2.5-mil Diametral Clearance, 1 Kr + 3 Xe Atmosphere, Production UO_2

△ WAPD 22-1: 3.5-mil Diametral Clearance, Helium Atmosphere, MCW UO_2, Laboratory Pressed and Sintered

▽ WAPD 22-3: 8.0-mil Diametral Clearance, Helium Atmosphere, Production UO_2

▲ WAPD 22-9: 8.0-mil Diametral Clearance, 1 Kr + 3 Xe Atmosphere, Production UO_2

▼ WAPD 22-2: 8.0-mil Diametral Clearance, Vacuum Atmosphere, Production UO_2

◊ WAPD 22-7: 13-mil Diametral Clearance, 1 Kr + 3 Xe Atmosphere, Production UO_2, Initial Startup

♦ WAPD 22-7: 13-mil Diametral Clearance, 1 Kr + 3 Xe Atmosphere, Production UO_2, Second Startup

Fig. II-5 Effective thermal conductivity of PWR production UO_2 in the WAPD-22 experiment.[17]

Several irradiation tests of PWR core 1 fuel-element specimens resulted in central melting of the oxide.[4] A few of these specimens had failures that were directly attributable to central melting; other specimens did not fail even though 42 per cent of the volume was molten. Since there appeared to be no basis for setting an allowable amount of melting, the thermal design limitation of "no central melting" was established[4] for PWR core 1. This thermal design criterion has become a standard for UO_2 fuel elements.

It is generally agreed that the poor thermal conductivity of UO_2 is the most serious disadvantage of this fuel. Paradoxically, however, the low conductivity of UO_2 brings into effect a very important transient shutdown mechanism in low-enrichment reactors. An increase in reactor power causes a large increase in fuel temperature because of the low conductivity of UO_2. As the fuel temperature is increased, Doppler broadening of the U^{238} absorption cross section causes an increase in nonfission absorptions and thus quenches the power excursion. Two General Electric reports[18,19] give an indication of how this shutdown mechanism acts to limit fuel-element damage during a startup accident.

Startup accident analyses were performed for three types of light-water-moderated reactors.[18] These reactors were characterized by their fuel elements as follows: (1) uranium-aluminum alloy, highly enriched, in aluminum-clad flat plates; (2) uranium-zirconium alloy, highly enriched, in zirconium-clad flat plates; and (3) UO_2, slightly enriched, in zirconium-clad rods. The purpose of the study was to determine whether a metal-water reaction could be initiated by a startup accident; molten metal was considered to be a prerequisite for a metal-

water reaction. The results indicated that, even with an initial period of 4.34×10^{-3} sec, the UO_2 fuel elements remained well below the melting temperature of zirconium, as shown in Fig. II-6.

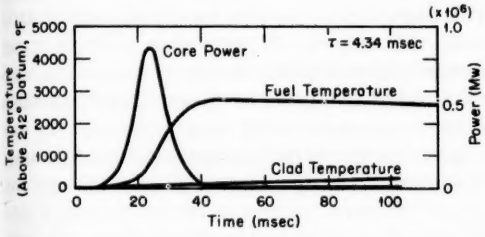


Fig. II-6 Typical power and temperature excursion for a zirconium-clad UO_2 fuel element.[18]

With the same initial period, the other two cores suffered extensive fuel-element melting. A stress analysis was then performed[19] on the UO_2 fuel elements using the temperatures obtained from the transient study. The conclusion was that the cladding would not rupture as a result of a startup accident with an initial period of 4.34×10^{-3} sec.

Of course, the results of this startup accident analysis are not generally applicable. However, it appears that the results are qualitatively correct, and the analysis demonstrates an inherent feature of UO_2-fueled low-enrichment reactors which is extremely important to fuel-element integrity, as well as to over-all reactor safety. (T. D. Anderson)

Reactor Stability Theory

The equilibrium power level of a nuclear reactor is that power level which will remain constant in time once it has been achieved. A nuclear reactor is said to be stable if the power returns to its equilibrium value after an arbitrary small disturbance. A reactor without explicit time dependence built into it is conveniently referred to as a stationary reactor and is an excellent example of the concept of an autonomous system, familiar in servomechanism theory. A stationary reactor will usually have at least one equilibrium power level, sometimes more than one, and perhaps (in rare and abnormal cases) none. This equilibrium power level depends, of course, on the various parameters of the complete reactor system and will change as the settings of the reactor controls are altered.

Reactor stability is by no means automatic, and it can be assured only by careful design. It is convenient to divide reactor stability theory into the cases of linear stability and nonlinear stability. The theory of linear stability is mathematically identical with linear servomechanism theory and deals with the question of the time behavior of a reactor with a power infinitesimally different from its equilibrium power. The theory of nonlinear stability, which attempts to describe the time behavior of a reactor with power initially arbitrarily removed from the equilibrium value, is mathematically much more difficult and, fortunately, less important in practice.

Although the classical nuclear excursion type of accident can be regarded simply as the approach to equilibrium power (with overshoot, usually) after a finite disturbance, it is conventionally not included in the subject of reactor stability. The question of whether the power approaches its equilibrium is irrelevant to the calculation of the total energy released. The relation of reactor stability theory to questions of reactor safety is rather through the prediction (and prevention) of accidents in which the departure of the reactor from its planned power level is initially completely nonhazardous in size. The development of the accident then proceeds by a steadily increasing (oscillatory or monotonic) departure from the equilibrium power. To be really dangerous, such unstable growth must be too rapid to be handled easily by gross safety provisions, such as the release of safety rods. Under these conditions, injury to personnel or damage to the reactor becomes possible, with meltdown being the typical end result.

Probably the most impressive accident of this type was the meltdown of the Experimental Breeder Reactor (EBR-I, Mark II). This reactor possessed a clear-cut negative power coefficient of reactivity (reactivity decrease with increase of power) and would therefore have appeared to be stable. Unfortunately, an instantaneous input of fission heat caused, at first, a reactivity rise, followed by a reactivity fall. At a sufficiently high equilibrium power level, this time dependence brought about a power oscillation of increasing amplitude that destroyed the reactor.

The general theory of reactor stability, with applications to a number of different reactor types, has been developed in the last 10 years. In addition to the application of Nyquist's theorem

and other standard methods of linear servomechanism theory to the problem of determining linear stability, several general theorems have been deduced concerning the effects of delayed neutrons and various other reactivity time lags on nonlinear stability. It was originally proved by Ergen and Weinberg[20] that the delayed neutrons always tend to cause a strongly damped return of the reactor power to its equilibrium level following an arbitrary disturbance. Welton,[21] who prepared this review, derived a criterion under which the other time lags (associated with reactivity changes produced by power variations) would similarly produce a damping effect. Although neither result has since been seriously questioned, there has been much uneasiness over the quasi-mechanical treatments used by these authors. It was thought, with some justification, that the formally elegant results should be derivable by somewhat less inelegant procedures, and these misgivings have in fact led to a number of interesting attempts to provide a suitable formal framework for the results.

The first of these was reported in a short article by Smets,[22] who finds it convenient to restrict himself to the case of no delayed neutrons, their effect being certainly to help ensure stability. In essence, he has found a criterion for the stability of the reactor by the use of Liapunov's second method, which is much more of a standard procedure than the mechanical analogy used by the reviewer.[21] He writes

$$\frac{dn(t)}{dt} = \frac{k(t)}{l}\, n(t) \qquad (1a)$$

$$k(t) = k_0 - \int_0^\infty du\; G(u)\; [n(t-u) - n_0] \qquad (1b)$$

where

$G(u)$ = the kernel which describes the time lags in the reactivity adjustment
$n(t)$ = the number of fissions per second (proportional to power)
$k(t)$ = the multiplication factor
l = the prompt-neutron generation time
t,u = time variables

Smets then notes that Eqs. 1*a* and 1*b* are equivalent to

$$\frac{d}{dt} N(t) = \frac{1}{l} \left\{ k_0 - \int_0^\infty du\; G(u)\; [n(t-u) - n_0] \right\} \times [n(t) - n_0] \qquad (2)$$

where

$$N(t) = n - n_0 - n_0 \log \frac{n}{n_0}$$

It is assumed that k_0 (the reactivity contribution from the control rod) has gone through some sort of variation for $t < 0$. The control rod is then set to make $k_0 = 0$ for $t > 0$, and $n(t)$ goes through a time variation which may cause $n(t)$ to approach n_0 for large t (stability) or to continue to deviate from n_0 (instability). Smets assumes that n behaves in such a way for large t that $\int_0^\infty dt\; [n(t) - n_0]$ is well defined; thus $n(t)$ may either approach n_0 in the stable case or oscillate with a limited or increasing amplitude in the unstable case. With this assumption, Eq. 2 can be integrated from zero to infinity, yielding

$$N(0) - N(\infty) = \frac{1}{l} \int_0^\infty dt \times \int_0^\infty du\; G(u)\; [n(t-u) - n_0]\; [n(t) - n_0] \qquad (3)$$

The function N has the properties

$$N > 0 \quad (n \neq n_0)$$

$$N = 0 \quad (n = n_0)$$

Thus, if $G(u)$ is such as to make the double integral necessarily positive, it is clear that

$$|n(\infty) - n_0| \leq |n(0) - n_0|$$

A similar argument could be made for any other initial time so that

$$|n(\infty) - n_0| \leq |n(t) - n_0|$$

and it then follows that $n(t) \to n_0$ for $t \to \infty$. Conversely, to obtain instability, the double integral must be capable of taking on negative values for some choice of $n(t)$.

This sufficient condition for stability is equivalent to

$$\int_0^\infty du\; G(u) \cos \omega u > 0$$

where ω is any positive, real frequency, which is precisely equivalent to the reviewer's criterion originally obtained by a much more complex argument.[21]

A somewhat more detailed treatment, with a very similar approach and identical results, is given by Akcasu and Dalfes[23] in a recent article. Both these authors and Smets make use of a familiar result from electrical circuit theory to obtain the equivalence which yields Eq. 3. In addition, Akcasu and Dalfes are able, by suitable use of Liapunov's second method, to demonstrate simultaneously the essential stabilizing effect of the delayed neutrons. It has long been an unsatisfactory aspect of the theory that the manipulations used by Ergen and Weinberg to demonstrate the delayed-neutron effect were incapable of demonstrating the full stability criterion. The method adopted by the reviewer[21] was, however, specially chosen for investigating the effect of "reactivity" time lags rather than "source" time lags and proved incapable of producing useful information on the delayed-neutron effect.

Another recent article on this subject is by Gyftopoulos and Devooght,[24] who attempt to reformulate the basic equations of reactor kinetics before studying the question of stability. In the reviewer's opinion their attempt at a more careful formulation leaves something to be desired in the direction of clarity and conciseness, but it does constitute a novel approach to an old derivation.

By a treatment which resembles the original one of the reviewer[21] more than do the first two described, these authors then show once more that stability is ensured by Eq. 3. Apparently, because of their attempted generality, they seem to have inadvertently ignored the effect of delayed neutrons in obtaining their stability criterion. Their methods do not in fact appear to be powerful enough to make this inclusion possible, and therefore on this point their treatment is actually comparable with that of Smets.

Another approach to the criterion for stability is exemplified in an article by Ergen et al.,[25] as well as a further article by Smets.[26] These authors make use of a rather special representation of the function $G(u)$ that is appropriate to the case of a reactivity which is a linear combination of a finite set of temperatures, θ_i, each governed by a first-order linear differential equation. Thus

$$G(u) = \sum_i G_i\, \theta_i(u) \tag{4}$$

$$\frac{d\theta_i(u)}{du} = \sum_i \lambda_{ij}\theta_j + \mu_i\, \delta(u) \tag{5}$$

where G_i is the coefficient governing change of reactivity with each temperature, λ_{ij} is a set of reciprocal relaxation times for the various possible heat-transfer routes, and μ_i is the coefficient that describes the effects of impulsive energy input at $u = 0$ on the temperatures of the various media of the reactor. The quantity $\delta(u)$ is the delta function having a value different from zero only for $u = 0$ and having a unit integrated value. Such a reactor model, although relatively more specialized than the case of general $G(u)$, is nevertheless a most useful description in practice. As the authors point out, the definitions given by Eqs. 4 and 5 allow Liapunov's second method to be directly applied to determine the restrictions on G_i, λ_{ij}, and μ_i to ensure stability. As expected, the criteria obtained agree precisely with those derived by specializing the reviewer's criterion to this model.

It may be properly argued that the general validity of the criterion of Eq. 3 is no longer in any reasonable doubt. A more profitable line of investigation would seem to be the study of the conditions under which the criterion is not satisfied but which are nevertheless perfectly well behaved. It is hoped that some useful reduction in the stringency of the conditions for stability can be obtained that is comparable with the precision of the results long available for linear stability.

One further article to be reviewed is a very detailed and careful study by Sandmeier[27] on the kinetics of fast reactors. Although the work is restricted to the type of fast reactors that have been extensively studied at Argonne as breeders, the work is so thoroughly done and so well described as to provide a very useful model for such investigations on other reactor types. The Argonne EBR-I, Mark II, reactor provides a beautiful illustration of the classical reactor accident arising from instability. The criterion of Eq. 3 was not satisfied, but the potentialities of the situation were apparently not understood until a series of power pulses of rapidly increasing size had largely melted the core. With such motivation, it is only natural that the kinetics of such reactors should have subsequently received most careful study.

A large part of Sandmeier's work is concerned with obtaining a reasonably exact theory of the linear kinetics of a reactor of the EBR type. He first examines the zero-power kinetics, which involves only the delayed neutrons. Thus

the function of principal interest is the so-called "zero-power transfer function":

$$ZP(s) = \frac{1 - s\sum_i \frac{\beta_i}{s + \lambda_i}}{s\left(l + \sum_i \frac{\beta_i}{s + \lambda_i}\right)}$$

where the variable s refers to a time dependence of the departure from equilibrium which is proportional to e^{st}. The parameter l is the prompt-neutron generation time ($\sim 5 \times 10^{-8}$ sec for a fast reactor), β_i is the effective fission yield of the ith delayed-neutron group, and λ_i is the associated mean life. The above function is directly measurable for values of s on the imaginary axis ($s = i\omega$) by the so-called oscillator test, which involves the production of a sinusoidal time variation of the departure from equilibrium power by means of an applied sinusoidal variation of excess reactivity. Thus, if $\delta n(t)$ and $\delta k(t)$ are assumed to have time variation $e^{i\omega t}$, it can be shown that

$$\frac{1}{n_0}\frac{\delta n}{\delta k} = ZP(i\omega) \qquad (6)$$

where n_0 is the mean power level, and the complex representation of the time dependence has been utilized for compactness.

For computational convenience, a somewhat simplified representation of $ZP(s)$ is found, and a study is made of the effect of $ZP(s)$ produced by the uncertainties in the interpretation of the delayed-neutron data. In general, Sandmeier concludes that these uncertainties are not of great importance and that a simple approximation to $ZP(s)$ is in fact adequate, a conclusion in which the reviewer concurs.

As the next step in the development of the linear kinetics, a discussion is given[27] of the various effects by which reactor power variation produces variation of reactivity. As in all reactors, these divide into two principal classes. The first class consists of variations in neutron cross sections arising directly from heating. In this class is the Doppler effect on the resonance absorption cross sections, which is of importance for fast reactors and is discussed by Sandmeier, as well as the change of mean energy for thermal neutrons, which has no relevance for fast reactors.

The second class consists of effects arising from thermal expansion. In this category are the simple increases in neutron leakage associated with thermal expansion of various regions of the reactor, as well as reactivity changes associated with distortions of the reactor structure produced by differential heating. The simple increase of leakage by thermal expansion is, of course, common to all reactor types and is the most dependable source of automatic reactivity control. Although reactivity change by distortion can occur in any type of reactor with fixed fuel elements, it has in fact been noticed only in the EBR type of reactor, where differential heating of the fuel pins causes an inward bowing of the pins, with consequent reactivity increase.

All these effects are complicated in a heterogeneous reactor by the time lags associated with the heat-transfer process. Thus a sudden increase of reactor power will produce an immediate rise of fuel temperature, with its associated Doppler effect and fuel pin bowing, but an increase in leakage must await transfer of heat from fuel to expansible coolant.

Sandmeier quotes the available work on the Doppler effect and concludes it to be relatively unimportant in the treatment of the kinetics. However, the fuel pin bowing and leakage increases from expansion are known to be of crucial importance, and he gives estimated values for the appropriate coefficients. The fuel pin bowing is very small, except for EBR-I, Mark II, for which Sandmeier quotes an estimated rise of reactivity of 2.0×10^{-3} for a power rise of 1 Mw-sec. The reactivity coefficient describing the leakage increase is quoted as -3.0×10^{-3} for a similar power rise. The net of these two effects is a negative power coefficient of reactivity, which would be expected to yield stability.

Actually, the above numbers must be combined with some estimate of the time dependence of the reactivity change following a sudden energy input to decide stability. Thus impulsive energy input to the fuel from fission will cause an essentially abrupt (neglecting inertial effects) increase of reactivity (from the fuel pin bowing) followed by a decrease of reactivity as the heat leaks out from the fuel to the flowing coolant. As the coolant heats up, the reactivity will further decrease because of the expansion of the coolant and the parts which it bathes. Finally, the reactivity will return to normal as the heated coolant flows out of the reactor. Clearly, the system possesses at least two relaxation times, although a description in terms of only two exponential decays can only be a rough ap-

proximation to the actual system. Sandmeier deduces a fuel-to-coolant relaxation time of 0.5 sec, and a coolant-to-external system relaxation time of 5.0 sec is assumed. This latter figure is actually not sufficiently discussed for clarity, but it presumably is of the same order of magnitude as the mean residence time of coolant within the fuel region.

An s-dependent power coefficient of reactivity is then defined as

$$K(s) = \left[\frac{2 \times 10^{-3}}{1 + 0.5s} - \frac{3 \times 10^{-3}}{(1 + 0.5s)(1 + 5s)}\right] \qquad (7)$$

where all the numbers are recognizable from their values as identical with the two power coefficients and two relaxation times already introduced. This rough analysis is typical of the sort of calculation which is often adequate to decide questions of stability. Clearly, a precise analysis is extremely difficult and probably unprofitable, in view of parameter uncertainties. It may be well to note the basis for Eq. 7. It is the Laplace transform of

$$K(\tau) = \frac{2 \times 10^{-3}}{0.5} e^{-\tau/0.5} - \frac{3 \times 10^{-3}}{5 - 0.5} \times (e^{-\tau/5} - e^{-\tau/0.5}) \qquad \text{for } \tau > 0 \qquad (8)$$

and

$$K(\tau) = 0 \qquad \text{for } \tau < 0$$

of which the first term describes the prompt increase of reactivity, following energy input at $\tau = 0$, and the second term describes the subsequent decrease of reactivity by heating of the coolant, with the eventual complete decay of the effect by coolant outflow.

A useful approximation is to assume that $K(s)$ is independent of the power level at which the reactor is being run. This is expected to be approximately valid because both the coolant flow rate and the fuel pin thermal conductivity will be essentially independent of power level. (The flow rate can, of course, be varied if desired.)

Sandmeier then writes a relation which is analogous to Eq. 6 but which takes into account the additional reactivity variation associated with δn through the coefficient $K(s)$:

$$\frac{\delta n(i\omega)}{n_0\ \delta k(i\omega)} = \frac{ZP(i\omega)}{1 - ZP(i\omega)\ PK(i\omega)} \equiv LP(i\omega) \qquad (9)$$

Again, this relates the amplitude and phase of sinusoidal flux variation δn to that of the externally controlled reactivity δk. Sandmeier's notation is somewhat confusing, ZP and LP being functions of s, and $PK(i\omega)$ signifies equilibrium reactor power in megawatts (P) multiplied by the power coefficient $K(i\omega)$.

For $P = 0$, the reactor response is determined completely by the delayed neutrons ($LP = ZP$); whereas, for increasing P, the denominator of Eq. 9 deviates increasingly from unity. If P is increased, and there exists some real value of ω for which $ZP(i\omega)\ K(i\omega)$ is a positive, real number, a value of P will be reached for which a finite δn is produced by vanishing δk. Any increase of P beyond this critical value then gives rise to an instability for which the reactor power oscillates about equilibrium with exponentially increasing amplitude and frequency, as defined above.

Sandmeier has shown in his first section that the phase angle of $ZP(i\omega)$ is between 0 and $-\pi/2$ for all real, positive ω, which corresponds to the commonly used result that the delayed neutrons necessarily contribute a stabilizing influence. It follows that instability can only arise if $K(i\omega)$ has a phase angle between $\pi/2$ and π for some real, positive value of ω. This is, in part, equivalent to the general stability criterion previously quoted in Eq. 3, $K(i\omega)$ being related to $G(u)$ by

$$K(i\omega) = \int_0^\infty du\ G(u)\ e^{-i\omega u}$$

It should be remarked, parenthetically, that Eq. 3 is a sufficient condition for stability, assuming only that the delayed-neutron source following fission is positive for all times. If, in addition, the delayed-neutron source decreases monotonically with increasing time, an additional condition on the imaginary part of $K(i\omega)$ can be derived to tighten up Eq. 3 and complete the correspondence with the condition on the phase angle of $K(i\omega)$ just given.

Sandmeier gives a very detailed set of calculations of the properties of $LP(s)$, assuming Eq. 7 for $K(s)$. For a power slightly in excess of 3 Mw, an undamped oscillation appears at a frequency of about 0.4 rad/sec, and a higher power will lead to a rapidly increasing amplitude. It should be emphasized, of course, that this result applies only to EBR-I, Mark II, since EBR-I, Mark III, and EBR-II were de-

signed to reduce the fuel pin bowing to a negligible effect.

Although the work is primarily on the linear kinetics of the reactor (certainly the most important consideration, by far), Sandmeier devotes a chapter to nonlinear effects. He calculates the Fourier coefficients of the flux waveform in terms of the coefficients for the reactivity waveform, this information being of considerable use in the interpretation of oscillator test results. Finally, he discusses another method for eliminating the harmful effects of fuel pin bowing; namely, by the use of a mechanical stop to limit the amount of bowing. He shows that the effect is simply to allow sustained flux oscillations of limited amplitude.

In general, Sandmeier's work is representative of the many detailed kinetic calculations which have been made in the last few years for the various important reactor types. They require a nice combination of physical insight, computational skill, and patience and have as their desired (and usually attained) end product a reactor that is free of kinetic difficulties. Many of the points which must be studied are subtle, and it is noteworthy that the reactor designers are able to prevent serious design blunders in almost all cases. (T. A. Welton)

Review of KEWB Experimental Program

The Kinetic Experiment on Water Boilers (KEWB) program[28–32] is an Atomics International project for the study of the kinetic behavior of the water boiler type of reactor. Water-boiler reactors are low-power homogeneous reactors with cores of aqueous solutions containing enriched uranium and having the following three characteristics: a small core vessel (less than 30 liters volume), a graphite reflector, and an external plumbing system for processing the gases evolved in the core. These reactors are of interest because of their simplicity and their large negative temperature and void coefficients which limit the transient excursions.[33]

The KEWB program, undertaken in 1954, was divided into three projects: an experimental study to obtain data, a program for the development of instruments suitable for measuring transient pressures and temperatures, and a theoretical study of kinetic behavior. The instrument development project has been concluded. The other two projects are now included in the Commission's reactor safety program.[34] The stated objectives of the KEWB studies are to provide design data to ensure maximum safety and to develop analytical methods for predicting the dynamic behavior of solution type reactors.

Two core vessels, designated A and B, have been used in the KEWB studies. The experiments using the "A" core vessel,[28–30] a 12.3-in.-ID stainless-steel sphere, have been completed. The fuel solution of uranyl sulfate occupied 11.5 liters of the total vessel volume of 13.6 liters. This loading contained 1905 g of U^{235} and had an excess reactivity of 4 per cent at 25°C. Stainless-steel cooling coils were included within this core so that the fuel temperature could be maintained below 80°C during operation up to the design power of 50 kw. The reflector, which was reactor-grade graphite bars, was stacked on a concrete pedestal to form a 56-in. cube that was enclosed by a gastight, reinforced, sheet-aluminum container. Core B, on the other hand, is a 12.3-in.-diameter cylindrical core, 36 in. high, which is loaded to an excess reactivity of about 5 per cent with 1770 g of U^{235} in 18.0 liters of fuel solution. Except where noted, the following discussion pertains to results obtained from experiments with core A.

Static Experiments

The experiments may conveniently be classified as static, periodic, or transient. The static experiments (references 30–33, 35, and 36) have included control-rod calibration (this is actually done by operating on so-called "stable" periods) and measurements of the reactivity coefficients which arise from temperature effects and from the radiolytic dissociation of water-forming bubbles of free hydrogen and oxygen.

For a core pressure of 68 cm Hg, the temperature coefficient of reactivity was found to be -0.016 per cent $\Delta k/°C$ at 25°C, increasing in magnitude to -0.027 per cent $\Delta k/°C$ at 85°C and to -0.032 per cent $\Delta k/°C$ at 90°C. The void coefficient of reactivity was determined using calibrated volumes in the horizontal central control tube. The result was -0.004 per cent $\Delta k/cm^3$.

The radiolytic gas production rate (measured in liters at standard temperature and pressure per kilowatt-hour) was independent of the reac-

tor power for power levels from 0.5 to 20 kw. This radiolytic gas production rate was also found to be independent of temperature for a core pressure of 60 cm Hg, remaining at the value of 14.0 liters/kw-hr in the temperature range from 30 to 90°C. Below 30°C there was a slight decrease to a rate of 12.9 liters/kw-hr at 17.5°C. However, the radiolytic gas production rate was shown to vary with the core pressure. The rate decreased linearly from 15.4 liters/kw-hr at 27 cm Hg to 13.7 liters/kw-hr at 68 cm Hg for a core temperature of 30°C.

Reports[37,38] of experiments on the cylindrical B core indicate results similar to those obtained with the A core. Four vertical control rods were precisely calibrated to permit the construction of an experimental inhour curve for the B core. As for the A core, an excellent fit to this curve has been obtained using the conventional six fission-product delay groups plus a seventh reflector delayed-neutron group. The same seventh-group parameters were used in the analysis of the B core as for the A core. Their use was largely a matter of convenience because of the limiting experimental uncertainties.

Periodic Experiments

The periodic experiments[31,39] measured the reactor power response (the phase lag, as well as the magnitude) to an oscillating reactivity variation. The oscillator was of the mechanical, rotary, modulator type and consisted of two, nested, coaxial cylinders inserted into the central exposure tube. The outer brass cylinder (the rotor) was rotated about the fixed inner steel cylinder (the stator). A sinusoidal reactivity modulation resulted from the shielding of sinusoidally shaped cadmium "spots" on the stator as they were swept across by small rectangular cadmium "shades" on the inner surface of the rotor. Four perturbation cycles occurred for each revolution of the rotor.

This oscillator technique allowed the experimental determination of the amplitude and phase of the transfer function in the frequency range from 0.022 to 260 cps. From this experimentally determined transfer function, the ratio l/β of the effective prompt-neutron lifetime to the effective fraction of delayed neutrons was calculated to be 7.7 ± 0.3 msec for all but the fastest transients.

The neutron lifetime for slow transients was determined to be 61.8 ± 2.4 μsec using an effective fraction of delayed neutrons of 0.008. This determination is to be compared with 62.5 μsec as determined from the experimental inhour equation and with 78.0 ± 3.0 μsec as determined from an analysis of random noise.

The transfer function indicated an effect similar to that which would result from a seventh neutron-delay group with an effective abundance of 0.023 and a mean delay time of 2 msec. This was interpreted to be due to those neutrons which are returned to the core from the reflector.[40,41]

Transient Experiments

The transient experiments[30,40,41] investigated the effects of the amount of reactivity release, rate of reactivity insertion, initial temperature, initial pressure, initial power, and initial void volume in the core. The excursion data were obtained from oscillograph recordings of the reactor power, both linear and logarithmic, of the core temperature, and of the pressures at the bottom and at the top of the core.

Transients were initiated by withdrawal of one or more of five B_4C control rods that controlled, altogether, a reactivity of about 12 per cent. Four control rods representing about 7 per cent of the activity entered the core through reentrant thimbles equally spaced about the vertical diameter. The fifth poison rod representing about 5 per cent reactivity was located in a central exposure tube along a horizontal diameter.

The transients were initiated by either step increases or ramp increases in the reactivity. For the step inputs the conditions at the initiation of each excursion were the following: (1) there was no flow of cooling water (this has little effect on the short transients, since the flow transit time is 0.55 sec); (2) the reactor gas-handling system was valved off from the core (the gases were collected in a ballast which was evacuated before initiation of the transient; an experimental investigation showed that the dynamic behavior is not significantly changed by including the gas recombination system); (3) the initial void volume above the core equaled 2.1 liters (except in the studies of the completely filled core); (4) the initial core temperature was adjusted to 25°C (except in the studies of the effect of the temperature);

(5) the initial core pressure was adjusted to one of the three values 15.6, 43, and 71 cm Hg.

The step-reactivity inputs were varied from 0.1 to approximately 4 per cent Δk and caused reactor periods ranging from 80 sec to 2 msec. The maximum power observed in the 2-msec period tests was 530 Mw with the under-full core and 780 Mw with the whole core. The maximum energy released for these core loadings was 3 and 6 megajoules, respectively.

Variations in the initial core temperature, initial pressure, initial power, and initial void volume above the core were found to cause only minor effects on the kinetic behavior of the reactor. That is to say, the peak power is determined mainly by the initial period of the excursion.

One of the most significant of the parameters varied is the initial void volume above the core solution. This volume was either 2.1 liters or 0. The expansion pressures in the filled core were two to four times the expansion pressures of the core with initial voids. Reflected pressure waves, which appeared in the core with voids, were absent in the filled core although the relative hazard was greater for the filled core vessel.

For a given initial period, the peak power decreases slightly with an increasing initial core temperature. Because of the increased water-vapor content of the gas bubbles, this decrease becomes more pronounced at the lower pressures and higher temperatures where boiling contributes more significantly as a shutdown mechanism.

An almost complete lack of influence of the initial power on the excursions was taken as evidence that the shutdown mechanisms are not affected by the initial presence in the core of gas voids at atmospheric pressure. The reactor peak powers were sensitive to the initial core pressures only for cores with initial voids on excursions with long periods (>0.2 sec). For these cores, higher initial pressures corresponded to higher peak powers.

In addition to step-reactivity increases, ramp-reactivity inputs, which increase linearly with time, were investigated. The reactor was initially set at essentially zero power with a core temperature of 25°C and with no flow of coolant. Reactivity was then inserted at 0.032, 0.063, 0.095, or 0.126 per cent Δk/sec until a total reactivity of 1.2 per cent or 2.4 per cent Δk was reached. The power rose rapidly and then fell, forming a sharp pulse. This was followed by a gradual power increase that was often accompanied by highly damped oscillations of small amplitude and cyclic periods of 2 to 3 sec. This increase continued until the ramp-reactivity increase stopped. Then the power approached an equilibrium value determined by the static reactor parameters and the total energy release. The excursions were observed for 1.5 min after reaching peak power, and then the reactors were shut down manually. It was shown by these tests that the peak power arising from a ramp input is very nearly the peak power from a step input of initial period equal to the minimum ramp period observed during the pulse.

In another experiment the reactor was operated for 30 min without cooling following a 2.4 per cent Δk ramp excursion, with the result that the power declined until the fuel solution reached boiling temperature, thereafter remaining constant at that level of a few kilowatts. This experiment is of interest in that it shows that there would be no serious consequences to accidental withdrawal of the control rods to 2.4 per cent Δk even if corrective reinsertion of the rods following their withdrawal was not possible for a considerable time.

Conclusions

The evidence of the KEWB experiments indicates that three important mechanisms[41] are involved in the reactor shutdown: (1) temperature, (2) radiolytic gas voids, and (3) reflector-delayed neutrons. Although the treatment of the reflector-delayed neutrons as a delay group avoids the difficult but more proper basic treatment of the problem, the concept has increased the understanding of short-period kinetic behavior. The reflector-delayed-neutron concept[41,42] explains, at least qualitatively, three anomalies associated with short transients: (1) the experimental inhour equation, (2) holdup of the power decay after attainment of peak power for short transients, and (3) the pile oscillator data. The reflector-delayed neutrons act as a reactivity storage mechanism that is especially apparent for periods near or less than the mean neutron lifetime in the reflector. This storage of reactivity is required to explain the transient data. The earliest models[40] attributed this storage to a "time delay for void formation." This delay, however, turned out to be a rapidly decreasing

function of reactor period. A later model postulated a rapid growth of gaseous voids from an unknown interaction between fission fragments and existing radiolytic gas bubbles. Now, however, the concept of reflector-delayed neutrons allows a more conservative model[41] for void formation based on the following three assumptions: (1) the void consists of radiolytic gas bubbles, (2) the bubbles are formed on preexisting nucleation sites (dust particles, molecular aggregates, etc.) in the fuel solution, and (3) the bubbles grow by diffusion. Calculations of void-compensated reactivities based on this model are in reasonably good agreement with the results of several transient tests.

Much useful information seems to have been obtained by the KEWB program, and progress has been made in the interpretation of the data. Even though many models have been proposed, there still remains a need for a better understanding of the shutdown mechanisms[41] so that a model capable of accurately predicting the kinetic behavior of water-boiler reactors can be constructed. (J. H. Marable)

Reactor Loop Experiment Hazards Evaluations

The term "loop" is applied to several types of cooled experimental systems, the more common being:

1. Gas- or liquid-cooled component irradiation facilities in which the coolant system may be single-pass or closed-cycle.
2. Gas- or liquid-cooled reactor fuel irradiation facilities with a single-pass or closed-cycle coolant circulation.
3. Irradiation facilities in which nonfissionable gases or liquids are circulated.
4. Irradiation facilities in which gases or liquids containing fissionable materials are circulated.

A loop used for fuel testing is usually built to duplicate the conditions to be encountered in a proposed reactor, and thus in most cases the loop presents the same hazards as those to be associated with the reactor, with the exception of criticality. Most testing reactors are built to accommodate more than one loop; thus it is not practical to control loop conditions by controlling the neutron flux of the test reactor. This limitation makes control of the loop operating conditions generally more difficult than control of the reactor, since temperatures must be more carefully monitored and must be controlled by regulating the coolant flow or by applying external heating or cooling. Since the loop conditions of temperature, pressure, coolant material, and fuel or hardware are greatly different from those of the testing reactor, a buffer zone must be provided between the loop and the reactor. Also, in loops designed to test untried fuels and components, provisions must be made for controlling the effects of failure of the test specimens. Containment and shielding are essential, and systems for decontaminating the loop coolant and components may be needed. Although loops used only for hardware testing may not involve hazards to the degree encountered in fuel-testing loops, they do involve the same types of hazards, particularly with respect to pressures, temperatures, and induced radioactivity.

As with reactors, the ultimate safety provision for a loop is the containment system. Highly reliable primary containment is, of course, necessary for all types of loops. If the results of failure of the primary containment would be only of a nuisance level, secondary containment might not be required. If, however, failure of the primary containment could result in extreme hazards to personnel or to the testing reactor, secondary containment would be essential. Such containment would be designed for protection in the event of the maximum credible failure of the primary system.

Since containment is the final resort in the control of loop hazards, other control methods must be incorporated into the design to minimize the possibility of the need for use of the containment. Such control methods include heater cutoff, reserve coolant supply, pressure relief, and reactor power reduction. The reliability of these safeguards and of the control system for normal loop operation is only as good as the instrumentation used. Therefore only proven types of instruments are used, and these are generally installed in duplicate or triplicate.

In discussing the specific hazards encountered in loop operation, it is perhaps best to present a résumé of in-pile loop hazards as a generalized introduction and then to review hazards analyses and operating reports for certain loops.

Radiation

The radiation hazards associated with loops are, in most cases, the same as those of reactors. The sources usually examined for determining shielding requirements are:

1. Direct radiation from the reactor through a gas-filled loop pipe.

2. Radiation from external pipes containing liquids or gases irradiated in the loop. (When fuel solutions are circulated or when there is the possibility that large fuel fragments may be blown from the loop into external components, shielding for delayed neutrons as well as for gamma radiation may become necessary.)

3. Radioactive gases, liquids, or particles released to the environment by rupture of the containment.

4. Radioactive parts ejected or accidentally removed from the reactor shielding or from transfer shields.

5. Radiation from specimens and components during removal, transfer, and maintenance.

Reactivity

Most reactivity hazards associated with loops are due to the influence of the loop on the testing reactor; however, some types of loops, such as those for testing homogeneous fuels, may build up critical inventories in dump or storage tanks. The usual methods by which a loop can influence the reactivity of the testing reactor are:

1. Collapse of voids, as by a gas-cooled loop becoming filled with water or pump oil.

2. Discharge of fluids into the reactor from pressurized piping. (The worst hazard in this case is the sudden stopping of the gas leakage after the reactor control system has adjusted to a loss of reactivity.)

3. Violent rupture of a high-pressure loop into a reactor tank. (In the ORR type of reactor, such an incident could destroy the reactor tank; in the MTR type of reactor, which has control rods suspended from the top plug and which has an expansion section in the reactor tank, such an incident could displace the top plug, and thus the control rods, in a direction to cause a sudden increase in reactivity.)

4. Sudden motion or ejection of the loop from the in-pile section.

5. Sudden removal or ejection of loop components from the in-pile section of the loop.

6. Leakage of fissionable or high-cross-section material from a loop into the reactor core or coolant.

Temperature

Heat is generated in loops from fission, from radiation and neutron energy absorption, and/or is supplied by external heaters. Since it is usually not feasible to use the reactor power to control the heating rate, other means, such as coolant flow control, position adjustment, input heat control, or thermal conductance control of insulating volumes, must normally be used. Reduction of the reactor power is normally used only as a last resort in reducing temperatures. Some of the situations leading to excessive temperatures are:

1. Failure of coolant flow without corrective action either by reactor power reduction, turning off externally supplied heat, or activating emergency coolant supplies.

2. Rupture of high-pressure systems so that coolant is lost. (Such incidents may cause meltdown of fuel specimens if the high pressure is required for adequate heat removal.)

3. Inadequate monitoring of specimens or loop parts so that melting or excessive corrosion results from lack of warning of excessive temperatures. (This condition could arise if sensing elements became inoperative during operation and loop shutdown were postponed.)

4. Insufficient coolant reserve to correct for allowable reactor excursions. (Most reactors of the MTR and ORR type are designed to correct only for a 10 to 20 per cent slow power excursion and for a 40 to 50 per cent fast power excursion if the servo control should fail. If the loop were not instrumented to require a reactor scram or power reduction in the event of an excessive temperature that might occur when the normal reactor power was exceeded, reserve coolant would be required.)

Pressure

The hazards of pressure effects on reactor reactivity, loss of coolant, and containment have been discussed under other topics. Some causes of high pressure are:

1. Temperature excursions.

2. Explosive recombination of radiolytic gases.

3. Failure of pressure-reducing valves from gas supply tanks and the resultant pressurization of the loop.

4. Leakage of water or other liquids into hot, low-pressure parts of the loop.

5. Downstream shutoff of coolant flow.

6. Long-term production of fission-product gases or volatile materials inside experimental capsules. (This, of course, will probably not affect the loop containment but will disperse radioactivity throughout the loop.)

7. Loss of insulation in systems cooled with liquefied gas.

8. Reaction of ozone with other materials in systems cooled with liquefied gas.

9. Rupture of any high-pressure part into a low-pressure part.

10. Presence of water or organic material in a sealed volume in a radiation field.

A different type of pressure problem arises when a low-pressure loop is operated in a high-pressure reactor. Collapse of any voids in the loop because of shell failure could cause a sudden increase in reactivity in some types of reactors.

Hardware and Materials

The hazards involving the loop hardware and other materials of the system include essentially all the loop hazards initiated by abnormal temperatures, pressures, and motions, since no hazard exists until the abnormal condition has affected the physical system. There are, however, some hazards that can be initiated by the hardware and materials as a result of improper design, poor fabrication, reaction to environment, and substandard materials. Failures have been or can be caused by:

1. Bad welds resulting from either poor techniques or the use of the wrong type of welding rod.

2. Porous pipe or porous containment walls.

3. Vibration resulting from insufficient bracing or lack of vibration suppressors, such as shock mountings and expansion joints.

4. Corrosion resulting from (a) chloride contamination in stainless-steel systems; (b) carbon, copper, or mercury contamination in aluminum systems; (c) use of a coolant that is not compatible with the structural materials (this may result not only in perforations but may cause pipe plugging because of mass transfer); (d) intermittent use of a facility with improper care of the environment between usages.

5. Radiation damage. (Service time for any system and for parts of the system should be predetermined so that affected parts can be replaced before radiation damage causes trouble.)

6. Poor design that does not allow for thermal expansion and/or creep so that warping and deformation can cause hot spots, fracture of internal parts, or rupture of the containment.

Components

The various components of a loop must, of course, be sized to handle their intended functions. In addition, they must have been proved reliable before being installed or fabricated into the loop complex. Since many loops operate at conditions which are extreme compared with those normally encountered in industry, special equipment must frequently be designed and built for the particular loop. Also, some standard equipment may be included and operated under conditions for which it was not specifically designed. If good pretesting procedures are not used for such equipment, failures may be expected. Since the best of any type of equipment, such as pumps, blowers, electrical relays, motorized valves, and thermocouples, is subject to failure, duplication or other types of safeguards are usually required to avoid serious trouble.

Instrumentation and Controls

Some loops are designed so that the instrumentation and controls problems are minimized. This is generally true of hydraulic- and pneumatic-tube irradiation facilities. In these loops the specimen size is kept small and the coolant flow large so that specimen instrumentation is not required. In most testing loops, however, temperature and pressure excursions cannot be tolerated without severe damage to the test specimens and/or the primary containment. The instrumentation and control system of the loop is, of course, the first line of defense against such damage. Generally, the loop properties that must be monitored and controlled by the experimenter for information purposes are those which must also be controlled for safety purposes. If the process of taking data does not compromise the safety action, then the same instruments can be used for both data and safety instruments. If the

safety action would otherwise be compromised, additional instruments must be provided to ensure safety action.

In furnishing a loop with safeguard instruments and controls, the degree of reliability of the systems used, as well as the nature of the hazards, should dictate the extent to which the safety action must be assured. Where the result of instrumentation failure may lead to severe hazards to personnel, to the reactor, or to large monetary investments, duplication of instrumentation and/or controls may be required to guarantee safety action. Duplication of safety instrumentation will increase the probability of a given safety action occurring when required but also may cause seemingly unnecessary reactor shutdowns when shutdown is the safety action required. At some installations, reactor operating time is considered sufficiently valuable to require triplication of instrumentation with two out of three instruments having to experience abnormal signals in order to cause a reactor shutdown or even a power reduction. (Editors' Note: Previous articles[43,44] in *Nuclear Safety* have discussed the implications of coincident and redundant instrumentation for control systems.)

Human Error

Probably the greatest diversity in causes of hazards lies in the realm of human error, which extends all the way from the design stage through fabrication, installation, operation, removal of the test specimens, and, finally, removal of the loop. Some of these errors are poor design and fabrication and the use of makeshift equipment. Human error might also be responsible for failure to develop safe installation and removal procedures so that there might be overloading of hoists, floors, and other equipment; use of inadequate shielding; inadequate radiation and contamination control; confusion in supervising the work; too many or too few people present; and inadequate handling equipment.

Further, carelessness during operation and maintenance could result in accidental bumping of switches, knobs, valves, and other equipment, with consequent adverse effects on safe operation; adjustment of the wrong controls; closing or opening of the wrong valves during sampling; lack of attention to oiling, changing batteries, standardizing, venting traps, and changing filters; and lack of attention to trends, such as an increase in flow resistance which might precede a sudden coolant stoppage due to scale formation and which might be followed by breaking away of chunks of scale. Lack of proper supervisory attention and administrative control might lead to a number of accident situations such as (1) the inadequate checkout of makeshift repairs, (2) jumping of safety-system relays to prolong operation, (3) attempting delicate repairs or adjustments during operation that should be done only during a shutdown, or (4) improper control of maintenance activities, such as chipping of concrete, which could cause large quantities of dust to accumulate on relays, or hosing of floors, which could lead to shorts in wiring.

Reactor Experiment Hazards Review

The purpose of reactor experiment hazards reviews is to eliminate, as much as possible, the elements of human error in the design, fabrication procedures, operating procedures, and final disposal of experiment facilities and experimental test assemblies. In order for such reviews to be accomplished, committees are usually established that are composed of members whose collective experience includes knowledge of all the types of hazards that can arise from operating any experimental system which they review. Formal committee reviews are at best, however, only superficial, and, in general, only major design, control, and operational shortcomings are detected. At ORNL, the hazards reviews of an experiment to be operated in an ORNL reactor begin when the plans for the experiment are first brought to the attention of the reactor operations groups.[45] Experienced operations personnel conduct a continuous review of these plans as they are developed by the researcher and the design engineers. Fabrication and installation are reviewed continuously by competent persons who see that each weld is made and tested properly, each flange is tightened exactly right, every sensor is installed properly, etc.[45] Immediately before startup, all components and controls are checked. Malfunctions or unexpected behavior in the loops which become apparent at that time frequently result in the loop startup being held over until the next reactor operating cycle, unless the corrections and adjustments can be made with only brief delay in the reactor startup.

The poor judgment that would lead to the operation of an expensive loop with known deficiencies in order to save time and money can be one of the worst types of human error. In order to have adequate checkout time, all the loop components should be ready for installation well before the reactor shutdown is scheduled. If possible, they should have been preassembled outside the reactor and checked for operability to uncover any hidden or unsuspected faults. Too tight schedules and last-minute assembling or repairs frequently result in substandard equipment and dangerous techniques. For example, a release of radioactive gases from a loop in the MTR occurred because a weld made during installation of the loop was not properly inspected.[46]

Operating Experience with In-Pile Loops at ORNL

A comprehensive description of nine types of loops that have been operated at ORNL, together with a discussion of failure experience, has been published.[47] The report[47] includes prescribed procedures for design, hazards evaluation, review, preoperational checks, and the setting up of operating standards. Of special interest are the discussions of reliability of safety instrumentation and other components. The one-out-of-two philosophy of safety action initiation is advocated. The advantages of having standardized instrumentation systems is stressed as providing the greatest reliability because of the greater ease with which design and maintenance personnel can provide and maintain such systems.

Eighteen aqueous homogeneous fuel test loops have been successfully operated at ORNL, and the accumulated experience has been reported.[48] Since many ruptures of high-pressure autoclave capsules have occurred in the reactor experiment facilities, it is remarkable that there have been no ruptures of loop pressure systems. Because the loops were operated under pressures and temperatures similar to those of the capsules (up to 2000 psi and 300°C), adequate secondary containment was provided for the in-pile portion; and, although no ruptures have been experienced, secondary containment is still a requirement for such loops. Similarly, external, vented, and shielded equipment chambers were used which could withstand a rupture of the pressure system. All components of each loop were checked before installation by operating the loop external to the reactor with the instruments for in-pile operation. The successful operation of these loops without serious incident has been due, to a large extent, to the elaborate, careful procedures worked out for each operation. Although as many as 50 sampling, filling, and control valves were installed on a loop, human error was essentially eliminated by the check-list type procedures used. A sample procedure is included in the report.[48]

WCAP-4 In-Pile Test Loop

The WCAP-4 is a high-pressure (2000 psi), high-temperature (600°F), water-cooled loop designed by Yankee Atomic Electric Company that is being operated in the MTR to test metal-clad UO_2 fuel specimens.[49] The in-pile section of the system is a hairpin loop with coolant flow down one leg and up the other. The hairpin section is enclosed in a vacuum jacket for insulating purposes. All the in-tank parts are made of stainless steel, except the lower portion of the vacuum jacket, which is made from an aluminum block to fit into the reactor reflector. The main external components are the coolant circulation pumps, an electrically heated pressurizer, a demineralizer bed, a degasifier, and the heat exchanger. In order to minimize leakage, the entire primary loop is of welded construction. External pressurized components are enclosed in a reinforced, shielded cubicle that is vented to the MTR gas disposal system.

Thorough analyses of temperatures, coolant flows, mechanical stresses, and shielding requirements were made prior to construction.[50] Time studies were also made to determine that a reactor shutdown could avert a meltdown in the event of stoppage of one or both coolant pumps. Before insertion into the reactor, the loop was assembled and operated to check coolant flows, expansion, effects of pressure, and the response of instruments and controls.

Instrumentation was provided in duplicate so that either of a pair could initiate a reactor power reduction in the event of high specimen temperature, high or low loop pressure, or loss of coolant flow. If spare thermocouples on the specimens were included, they were not mentioned; however, if no spares exist, loss of a thermocouple would mean termination of the test or operation with substandard safeguards.

If the in-tank portion of the pressure tube should rupture, the insulation shell would probably not serve as a secondary containment, since it was designed to withstand a pressure of only 100 psi. No analysis was made of the effect of rupture of the high-pressure system into the reactor or the equipment cubicle and the subsequent meltdown of the fuel specimens, since it was assumed that (1) this accident was not credible because of the type of construction and (2) the low-pressure safeguard would scram the reactor. Consequences of discharge of the volatile fission products into the MTR off-gas system were not considered, although this could occur if the rupture disk in the cubicle blew out from overpressure.

Molten-Salt-Fueled Loops Operated in the MTR

Descriptions and histories of the five ORNL-designed molten-salt-fueled loops that have been operated in the HB-3 facility at the MTR have been published.[46] The containment provided with this facility was comparable to the containment for the ORNL aqueous homogeneous fuel loops. It was compartmented and water cooled, and each compartment was individually monitored for the appearance of fission-product gases. Fuel circulated only within the in-pile portion of the facility, and the external components consisted only of an oil pump to drive the hydraulically operated fuel pump, various purge-gas systems, refrigerated charcoal traps, and instrumentation.

All safety instrumentation was double-tracked and designed to reduce the reactor power in the event of loss of fuel circulation, loss of cooling, high loop temperature, fission-product leakage, and failure of the system leak-detection purge-gas flow. Also, instrumentation was provided that would scram the reactor in the event of loss of electric power or instrument air.

The loop design, methods of fabrication, and safety instrumentation were reviewed independently by the ORNL Reactor Experiments Review Committee and by the equivalent MTR committee. Thus the thinking of both of these experienced groups, as well as that of an experienced design group, was reflected in the loop system. The reviews are reflected in the fact that in the five experiments the only release of radioactive gases was caused by failure of a weld made at the MTR which did not receive standard inspection due to expediency.

(C. D. Cagle)

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Section III

INSTRUMENTATION AND CONTROLS

*Use of 17-4-PH Stainless Steel for Critical Reactor Components**

By C. G. LONG†

The failure[1] in Commonwealth Edison's Dresden reactor of a control-rod index tube fabricated of 17-4-PH stainless steel and the detection of cracked tubes in other drive units of this reactor have resulted in a comprehensive review of the use of this material for critical reactor components. This material has been utilized in many reactor components where high strength and high hardness are required. The Dresden experience indicates, however, that under certain conditions this material is susceptible to cracking.

The Dresden failure originated from a circumferential crack that propagated to complete severance of the hollow index tube. Upon removal of the remaining drive units, cracks were detected in other index tubes and in other 17-4-PH steel components of the control-rod-drive system. These cracks were longitudinal, transverse, and circumferential, with some originating in the heavy sections and others extending from drilled holes. It is interesting to note that the index tubes had deep circumferential grooves machined approximately every 9 in. along their 10-ft length, but no indications of cracks were detected in or near these grooves. As a result of considerable testing and study, the General Electric Company concluded[1] that the cracks were caused by stress corrosion and that the material evidenced high residual stresses which had resulted from the fabrication and heat-treatment procedures.

At about the time of the Dresden tube failure, the 12 control-rod drives that had been in use for approximately 2½ years in Argonne's EBWR were removed from subsequent use in the new BORAX-V reactor. During ultrasonic examination of the 12 used shafts and two spare shafts, which had never seen service in the reactor, cracks were detected at the rod-drive end of one of the used shafts. This shaft was sectioned, and circumferential cracks up to 0.013 in. deep were discovered. In addition, one of the spare shafts was randomly sectioned, and several cracks up to 0.002 in. deep were found. No definite conclusion as to the cause of the cracks has been reached at Argonne; however, stress corrosion or the grinding operation performed on the shafts is suspected.

Concurrently, during a routine reactor shutdown, the rod-drive shafts of the Vallecitos Boiling-Water Reactor (VBWR) were inspected using dye penetrant. This inspection revealed[2] that one shaft had a series of longitudinal cracks in the hollow shaft adjacent to a shrink-fit mechanical joint. During reactor operation this shaft was subjected to the primary steam environment, and the cracking initiated from the inside wetted surface through the tube wall to the outside surface. General Electric concluded[2] that these cracks were also the result of stress corrosion. The shrink fit had placed the mate-

*The views expressed in this article do not necessarily represent those of the U. S. Atomic Energy Commission or its appropriate office or division.

†Charles G. Long is a member of the Research and Power Reactor Safety Branch, Division of Licensing Regulation, USAEC. He received a mechanical engineering degree from the University of Pittsburgh and has taken some graduate training at Pennsylvania State University. Before joining the AEC staff in 1960, he was employed as a Project Engineer for the Research Division, Curtiss-Wright Corporation, where he participated in the design and construction of research- and power-reactor systems and Project Pluto. He served with the U. S. Air Force from 1948 to 1952.

rial in a state of high stress and thus had made it susceptible to stress-corrosion cracking.

In view of the Dresden failure, the Division of Licensing and Regulation, USAEC, initiated a comprehensive review of the use of 17-4-PH steel for critical reactor components. Three expert metallurgical consultants from Argonne and ORNL aided AEC in this review. Information was obtained from reactor operators, reactor contractors, and the steel producer relative to the application, fabrication and heat-treatment procedures, inspections, test program results, and metallurgical data concerning the material. In addition, the results of nondestructive and destructive examinations of components made of 17-4-PH stainless steel which had seen extensive service in reactors were obtained for review and evaluation.

All the reactors in which failures and cracks had occurred utilized 17-4-PH stainless steel in their rod-drive shafts and in other components of the reactor systems. The heat-treatments employed to give the failed or cracked parts high hardness and strength were in the range 875 to 925°F. Based on the combination of possible high residual stresses, applied stresses, and service-water conditions, the general conclusion was reached that stress-corrosion cracking was the most likely mechanism of failure.

The two operating reactors, Dresden and VBWR, were studied, and numerous tests were performed by their designers to develop a suitable procedure for correcting the material problem. For Dresden, where high hardness and high strength were necessary, it was decided[3] by the plant designer to replace all the 17-4-PH stainless steel in the rod-drive mechanisms that required high hardness with 17-4-PH steel heat-treated to 1100°F. A detailed heat-treatment and inspection specification was followed, and it provided for nondestructive testing throughout the fabrication process and also prescribed only minimum straightening and a final heat-treatment after all straightening operations. Other 17-4-PH parts were replaced by parts fabricated of type 304 stainless steel and Inconel X. The material problem for the VBWR was resolved[2] by replacing all 17-4-PH steel parts, except one, with type 304 stainless-steel parts. The one remaining 17-4-PH part had functioned well during service, and dye-penetrant examinations disclosed no surface cracking. Furthermore, failure of this part would not impair the ability of the rod drive to scram.

Although the evaluation by the Division of Licensing and Regulation is not yet completed, particularly in regard to applications where chrome plating is required, certain tentative conclusions may be drawn. Generally, it has been concluded that the material problems experienced in Dresden and other reactors do *not* indicate that either the use of 17-4-PH steel regardless of heat-treatment or the use of 17-4-PH steel heat-treated in the 900°F range should be eliminated. The Shippingport reactor (PWR) has successfully utilized 17-4-PH steel in the 875°F heat-treated condition in the rod-drive lead screws since 1957. Recently, dye-penetrant inspections were made of PWR lead screws which had been removed from the reactor, and no indications of cracks or other abnormalities were found.

There are, however, certain facets of the problem that must be considered in attempting to compare the successful use of the material in one application with the proposed use in another. The raw material procurement and the fabrication procedures are quite often different with respect to material specifications, heat-treatment processes, quality control, straightening and finishing procedures, and final inspection methods. Furthermore, the service environment plays a major role in creating conditions that could induce, for example, stress corrosion or hydrogen embrittlement. In addition, some applications require chrome plating of the steel, which by itself might affect the surface of the base metal during the plating process by inducing cracks or embrittling the surface and thereby create conditions for eventual stress corrosion or load failure. Further, some reactor designers have specified shrink fits, interference fits, and other types of stress-inducing procedures that can place the material in a highly localized stress condition and thereby establish one of the conditions involved in stress corrosion. Finally, the threshold stress necessary to place 17-4-PH material in a susceptible condition for stress corrosion is not clearly known. Accordingly, the service application, with its normal and transient applied loads, should be carefully studied so that the resulting component design will be such that critical components will be exposed to minimum stresses during reactor operation. The factors discussed above explain why 17-4-PH stainless steel can

be utilized in one specific application with success, whereas in another similar but different application it may crack or fail.

Some specific steps that should be considered by reactor designers to help prevent failures of 17-4-PH components by stress corrosion or other means during service are as follows:

1. Detailed specifications for procuring raw material should be established and should cover chemical analysis, mechanical properties, and tests for surface and subsurface imperfections.
2. Proper heat-treatment procedures should be selected to obtain the desired mechanical properties.
3. The material should not be severely cold worked in any condition.
4. Any straightening after fabrication should be followed by a final heat-treatment, if possible. The use of jigs during heat-treatment should be considered.
5. Fabrication operations that could induce high residual stresses should be avoided.
6. Periodic nondestructive inspections should be performed throughout the fabrication cycle.
7. If chrome plating is required, precise procedures should be specified, including a final baking to drive off hydrogen.
8. Grinding operations should be strictly controlled to prevent damage to the material surface.
9. For critical reactor components, the material should not be used in the solution-treated condition.
10. It should be established by detailed engineering analyses and tests that the material with its as-fabricated properties and with the expected applied and transient stresses is compatible with the prospective service environment.

As mentioned above, 17-4-PH steel heat-treated in the 875 to 925°F range has been eliminated from some reactors, and the designers of these reactors have elected to refabricate the required parts using 17-4-PH steel heat-treated to 1100°F. Although experiments have indicated that the 1100°F heat-treatment reduces the possibility of occurrence of stress corrosion and for this reason the use of higher heat-treatment temperatures is advantageous, it must also be realized that the resulting material has lower hardness. For this reason the use of 1100°F heat-treated material must be critically examined to assure compatibility with the service conditions expected from a mechanical standpoint, particularly if chrome plating is required. A particular heat-treatment temperature alone is not the complete or necessarily the correct solution to the problems which may arise from the use of 17-4-PH stainless steel for critical reactor components, since many other factors may be involved. The acceptance of the use of 17-4-PH stainless steel for critical reactor components depends primarily on the thoroughness of material and inspection specifications, fabrication and test procedures, and compatibility of the design of the finished product with service conditions of stress and environment. (C. G. Long)

Instrumentation for Containment

Power and research reactors are provided with containment enclosures so that, even if the improbable maximum credible accident should occur, no one outside the controlled area would receive a serious radiation dose. Although the containment enclosure is usually considered to be very nearly gastight, it is in fact necessary to pierce it in many places for essential services to the plant within the vessel. Examples of such penetrations are water and steam lines, electrical conduits, and ventilating ducts. Each penetration must be provided with some means of sealing it so that, in case an incident should occur, the containment enclosure could then fulfill its function. Electrical connections may be sealed permanently, but the seals on pipes and ducts must take the form of valves which remain open during normal plant operation but which must be closed to make the containment effective. The instrumentation and control system which effects this valve closure is the subject of this discussion.

It is clear that containment valves are required to close when they are needed. Typical containment enclosure leakage-rate specifications range from 0.1 to 2 per cent per day, and even a relatively small valve which by failing to close may provide a path directly between the enclosure interior and the atmosphere will leak at a rate very much larger than specified. The case for valves in water and steam lines is not so clear. In some cases these lines lead to closed systems whose integrity may exceed that of the containment enclosure. Under normal conditions, transport of radioactivity into these

closed systems does not constitute a hazard to the general public. However, the auxiliary system might not be in its normal condition, and in any case it is certainly desirable for fluids which might be highly contaminated to be confined to the containment volume.

As an extreme example, consider a once-through air-ventilation system which normally exhausts through a stack. On the occurrence of an accident, valves are to close in the intake and exhaust ducts of the system. If the valve in the intake duct failed to close, any excess pressure in the containment enclosure could be vented directly to the atmosphere through the open intake port. The containment would thus be violated, with activity release perhaps taking place at or near ground level rather than through filters and up the stack. It is worth noting that it might be impossible to close such a valve by direct manual operation because of the high radiation level.

Another example of a valve vital to achieving containment is the vacuum-relief valve. Most containment vessels are built to withstand a positive internal pressure and are not braced for a negative internal pressure. Relief valves are therefore provided so that outside air can enter if needed to avoid a condition of excessive vacuum. These vacuum-breaker valves are not always shown in the piping drawings which are scrutinized to inventory penetrations of the containment enclosures. Such precautionary measures as are applied to most containment valves may therefore be missing from the vacuum-relief valves. It is certainly true, however, that a leaking relief valve can violate the containment just as the open ventilation-intake valve can.

Accidental closure of a containment valve when it is not needed for containment may also have serious consequences. In the first example given, accidental closure of the intake valve would interrupt ventilation. This might result in overheating of some plant components or in exposing personnel within the vessel to radiation from airborne activity that would normally be exhausted up the stack. Accidental blocking of steam lines, feed-water lines, or cooling-water lines could compromise major items of plant equipment. By appropriate design it has been possible to make some plants operate without outside ventilation. For these plants, of course, the problem of accidental closure of ventilating duct valves does not occur. In all the plants considered in this study, however, the energy is transferred from the reactor to an external turbine-generator by a hot fluid in a pipe that penetrates the containment enclosure. Therefore the closing of the containment valves would, at the least, interrupt power generation. It is therefore evident that containment valves are required to close every time the prescribed emergency arises. Furthermore, they should not close at any other time, since their accidental closure could result in at least a reactor shutdown if not a more serious thermal transient throughout the plant.

In nuclear processes, systems which accomplish a required control action with great reliability when action is needed, but which are guarded against inadvertent action, are called safety systems. A great deal of attention has been given to the safety system that decreases the reactivity if the reactor power exceeds a safe value. Generally, a substantial number of safety rods are provided so that complete failure of any one device will not significantly decrease the safety potential of the shutdown system. Similarly, several redundant channels of instrumentation are provided so that an instrument breakdown will not render the system inoperative. In addition, the outputs of the instruments are often combined, using coincidence techniques, to minimize the effects of a single danger signal, presumably false, which otherwise would initiate an unnecessary shutdown. The instruments used in this service have been highly refined and are built to more rigid, precise specifications than are analogous devices in, for example, chemical plants. The safety actuators (rod drives) are also designed with great care because of the exacting requirements of this safety-system service. Although it may not be so designated, the system comprising the containment valves and the instruments that initiate their closure is in fact a safety system.

It might indeed be argued that, since the containment safety system will probably not be needed during the life of the plant, it may be designed with less reliability than is required for nuclear safety systems. In the reviewer's opinion this reasoning is false. The containment enclosure is provided, often at great expense, to protect the general public outside the purview of the reactor operator against the consequences of the maximum credible accident. Correct action by the containment safety sys-

tem is essential to achieving containment, and it is therefore essential that the system operate reliably.

The requirements of the containment safety system are, in at least one sense, more stringent than those of the nuclear safety system. The nuclear process and its control may usually be taken as a single problem, and therefore the provision of a multiplicity of safety rods gives safety in numbers. Increasing the number of rods decreases the effect of the failure of any one of them to release, and the effect of single failures can be rendered unimportant in this way. In the containment safety system, however, the multiplicity of pipes and ducts to be blocked constitutes danger in numbers, since failure to block any single exit can violate the containment. The analogy to multiple safety rods lies in multiple valves to block each line. Provision of valves per line in numbers comparable to safety rods could be very expensive, especially for the case of large ventilating ducts where the extra safety is most needed. Since increasing reliability by buying safety in numbers is uneconomic, it is clear that the relatively small number of valves actually provided in each line must be as reliable and trouble-free as possible.

Certain features of the containment safety systems for eight power reactors in the United States are described in Table III-1. This information has been obtained principally from private communications[4–11] from the various project personnel, augmented in most cases by personal observation by the reviewer. This has been made necessary because information about the containment safety systems in Hazards Summary Reports is not usually sufficiently detailed to permit an evaluation such as is attempted here.

Table III-1 begins with a listing of the containment valves. In the different plants they are grouped in different ways; the most obvious dichotomy would separate pathways which could lead to external contamination from those which could not, but this is not always followed. In the table, *manual* and *automatic* both refer to a remotely operable valve; *local* refers to a valve that can be closed by a handwheel or other manual operator at the valve itself. The enclosure ventilation is considered separately since it is such a direct path for potential contamination leakage. It should be noted that the Yankee, Indian Point, and N.S. *Savannah* plants normally operate with their ventilation systems sealed.

Table III-1 also describes the action of the containment valves upon loss of electric power and instrument air. These descriptions are followed by a listing of the signals, or parameters, that must be sensed in order for the valve actuators to receive a signal to initiate closure. The number of sensors per parameter and the system logic, taken together, permit evaluation of the effects of component failure on system operation in terms of safety action when needed and of unnecessary shutdowns.

The tabulations of signal and valve monitors and possible emergency valve action pertain to the possible situations in which a human operator might be in a position to take action. In order for him to act, he must have the knowledge that action is needed and the means to effect the action.

From a study of the table, it is evident that some containment safety systems are not designed to have the same reliability as neutron safety systems. Although three independent instrument channels are considered standard for neutron flux or reactor period, many containment systems rely on fewer than three. The techniques of redundancy and coincidence are used in only one-half of the systems studied. In almost all cases a single electrical or pneumatic signal is sent to all valves; failure of the signaling device or of its energy source results either in automatic closing of all valves or in inability of any of the valves to close. Such a situation in a neutron safety system is now widely recognized to be poor practice. A large body of experience with false scrams has testified to the need for more sophisticated techniques to ensure that such a system will act when it is required and that it will not act spuriously. The apparent lack of such criteria in the design of some containment safety systems indicates that these systems should not be expected to give the same trouble-free service as modern neutron safety systems. Since correct action by the containment safety system is essential to protect off-site personnel (the general public) in the event of a nuclear incident, it is incumbent upon the nuclear power industry to examine critically the criteria on which the design of such systems is based. The first nuclear power plant incident will be the first real test of a containment system. It would be intolerable to discover that containment had been

Table III-1 FEATURES OF SOME REACTOR CONTAINMENT SAFETY SYSTEMS

Reactor	Number and Type of Valves per Line[a]			Action of Automatic Valves on Loss of Power	
	Vacuum Relief	Enclosure Ventilation	Process Lines[b] (Air, Steam, Water, etc.)	Loss of Electric Power	Loss of Instrument Air
VBWR	None	1 automatic, 1 local, normally open	1 automatic and 1 local in lines constituting possible routes for external contamination 1 local in other process lines	All valves close	Enclosure ventilation valves close
PWR		2 automatic, normally open	None in hydraulic valve lines 1 manual in main steam and feedwater lines 1 manual or local and 1 check valve in other inlet lines 1 manual or local in other outlet lines	Enclosure ventilation valves close (Note: cannot close enclosure ventilation on loss of hydraulic pressure)	Does not apply
Dresden	2 enclosure ventilation valves open automatically	2 automatic, normally open	1 manual and 1 check in inlet lines 1 automatic and 1 check in outlet lines	Motor-operated valves switch to station battery Other valves close	Valves close
N.S. Savannah	None	1 manual, normally closed	1 check valve in inlet lines 1 automatic in outlet lines; most lines have additional local	Close	Close
Piqua	1	2 automatic, normally open	1 automatic and 1 local in organic coolant lines 1 manual and 1 local in feedwater lines 1 automatic and 1 local in waste lines 2 local (1 inside, 1 outside enclosure) in other lines	Close	Close
Enrico Fermi	2	1 automatic, normally open, except none in closed-circuit nitrogen cooling loops	1 manual, locked closed, in purge line; other lines have 1 automatic normally open valve (Note: no water or steam lines in enclosure)	Close	Close
Yankee	None	1 manual, normally closed	1 automatic and 1 manual in main steam lines 2 check valves in other inlet lines 1 trip valve in other outlet lines	No electrically operated valves	Close
Indian Point	2	2 automatic, normally closed, in main system; 2 automatic, normally closed, in auxiliary system	2 automatic in lines constituting possible routes for external contamination 2 manual in vital service lines 1 automatic and 1 manual in other lines	Enclosure ventilation valves and valves in lines constituting possible routes for external contamination close	Close in lines constituting possible routes for external contamination

[a] Automatic indicates a valve closed by instruments. Manual denotes a valve operated remotely by an operator in the control room. Local means a valve operated by hand at or near the valve.

Parameters Sensed to Close Automatic Valves and Number of Sensors Per Parameter	System Logic for Automatic Valves	Monitors in Control Room		Possible Emergency Action if Valve Does not Close
		Signals	Valves	
Varies with valve, but includes 2 to 10 of following: High steam radiation, 1 Loss of power, 1 High stack radiation, 1 High condenser pressure, 1 Low reactor water level, 1 Low circulating water pressure, 1 High steam flow, 1 High enclosure pressure, 1 Seismic disturbance, 1 High enclosure radiation, 1 Manual operation possible	Any 1 signal to close valves	Individual lights	Individual lights	Close local valves, handles outside enclosure (also can be closed manually from control room)
High enclosure pressure, 2 High stack activity, 1 High enclosure air activity, 1 Manual operation possible	Any 1 signal to close enclosure ventilation valves only	High enclosure air activity only	Individual lights, enclosure ventilation only 1 annunciator	Close enclosure ventilation valves locally; would take considerable time
Enclosure ventilation valves, all scram parameters, 4 or 6 Other valves: High enclosure pressure, 4 Low reactor water level, 4 Manual operation possible	2-out-of-4 or 2-out-of-6 to close enclosure ventilation valves 2-out-of-4 for other valves	Annunciates only upon high radiation level or steam leak in enclosure	Individual lights and annunciator only on enclosure ventilation valves	Close manually from control room and, in some cases, locally also
High enclosure pressure, 3	2-out-of-3 to close valves	None	Individual annunciators	Close additional local valve where it exists
All scram parameters, 3 High stack activity, 3	2-out-of-3 Any 1 to close valves	Individual lights and annunciators	Individual lights for enclosure ventilation valves; only temperature, pressure, etc. indicators for other valves	Close locally (would take considerable time)
High particulate activity in enclosure, 4 High enclosure pressure, 3	Any 1 of four sensors to close all valves; three additional sensors close certain selected valves	Group annunciators	Individual lights	Close manually from remote station; some also close locally
High enclosure pressure, 2	Any 1 to close	1 annunciator	None	Close locally
High enclosure pressure, 8 High boiler water radioactivity, 4 Manual operation possible High stack radioactivity, 1 Low pressure level, 2	4-out-of-8 to close Each sensor closes valves isolating its monitored boiler Closes only enclosure ventilation valves 2-out-of-2 to close all valves	Individual lights and annunciators	Individual lights	Close locally; valve handles are in shielded room

[b] In this tabulation it has not been possible to take into account the pressure rating or special conditions that may apply to the systems to which these lines are connected.

ineffective because of instrument or control failure.

An outstanding feature of the information presented in Table III-1 is the great diversity of the approaches taken to assure containment. There are systems that initiate valve closure on 10 different parameters and systems that use only one. The number of valves installed per line range from two, automatically actuated, to none. Some of this variation can doubtless be ascribed to differences between plants and to the different potential hazards in the same plant, but in the reviewer's opinion the information in Table III-1 is indicative of the fact that the art of designing containment safety systems is in its infancy. The situation may be compared with that of nuclear safety systems 10 to 12 years ago, and it is to be hoped that, as in the nuclear safety field, further study of the basic problems will lead to general acceptance of suitable criteria for the design of containment safety systems. (S. H. Hanauer)

Control-Rod and Control-Rod-Drive Difficulties

There have been several reports of mechanical failure of reactor control rods or control-rod-drive mechanisms. Since control rods must have inherent ability to alter reactivity by large amounts, their failure to perform in a normal manner can often lead to hazardous reactor operation. The following reviews the reported difficulties with rods and drive mechanisms and presents comments on the safety implications.

One type of control-rod failure was reported by the operators of the Ford Nuclear Reactor at the University of Michigan.[12] In this particular case, swelling of one of the shim safety rods resulted in sticking of the rod in the special fuel element which served as the lower guide for the poison section. The binding was discovered only when it became impossible to withdraw the rod by normal procedures during a routine startup operation. Subsequent study led to the conclusion that the swelling was due to radiolytic dissociation of water that leaked into the flattened aluminum tube containing boron carbide powder. New rods were installed, and the reactor was subsequently operated for 300 Mw-hr. Upon removal of the new rods, measurements showed swelling of all three rods; also, one rod was off-gassing at a weld. The off-gases were analyzed, and the principal constituents were found to be hydrogen, oxygen, and nitrogen. Further investigations are under way.

Another case of rod binding occurred in the Brookhaven Medical Reactor, with the result that a rod failed to scram[13] in March 1960. There was no report of previous difficulty of this type. Swelling was found when the rod was removed at a later date. It is thought that internal helium gas pressure caused the deformation, although no gas analysis had been made at the time of the report.

Although no accident occurred as a result of either of these failures, it is quite obvious that potential hazards existed. The swelling of one or more control rods in a reactor could lead to jamming of the rods during operation and a reduction in the effectiveness of the safety system in the event of an accident. There is also the possibility, in some installations, of inadvertent withdrawal of a fuel element from the core by the swollen control element. Subsequent accidental release of the fuel could initiate a serious nuclear excursion. The AEC in recent letters to several reactor operators[14] has requested information relative to this type of accident. A third concern, in the case of the Ford Nuclear Reactor, is the possibility of detonation of the hydrogen-oxygen mixture in the shim safety rods. Although it appears that such an occurrence is improbable, the need for further investigation along this line has been mentioned.[12]

Another type of failure occurred in the Dresden reactor and was discovered[15] on Nov. 14, 1960. In this case a control blade (one of 80 in the reactor) became separated from its drive. The blade and drive were removed after the reactor was shut down. Investigation showed that stress-corrosion cracking had caused a complete break in the moving hydraulic cylinder which actuated the control blade. The decision was made to remove all 80 of the drives for examination and repair. The examination disclosed defective parts in 17 of the drives. Metallurgical studies indicated that the trouble could be eliminated by changing the heat-treatment of the stainless steel used in the rod-drive components (see the article on "Use of 17-4-PH Stainless-Steel Control-Rod Drives," this issue). In addition to the change in material, design improvements (principally in the buffer used for deceleration during scram) are being

made. A thorough test program is in progress to verify the suitability of the changes of the rod-drive mechanisms.

Separation of control rods from their drives during operation of the EBWR was reported by Argonne National Laboratory.[16,17] Two different cases were described. In one case a spot-welded transition section separated. This failure was attributed to dynamic forces imposed during rod oscillation to obtain power transfer functions. The second case was a partial failure of a riveted transition section. The rod never became separated from the drive, but the rivet failure allowed the drive to wedge in the core shroud guides. The cause of the rivet failures is being investigated.

The type of failure in which the poison member becomes detached from its drive can lead to a reactivity excursion, the consequences of which will be dependent on the amount of reactivity involved. Such an accident may occur if the physical arrangement is such that there is a credible mechanism for moving the poison out of the reactor without movement of the drive. In reactors wherein the control members scram upward against gravity, as was the case in Dresden, were a control element to become separated from its drive due to binding, etc., the drive may be withdrawn without removing the poison. If this should go undetected, it is possible that at some later time the faulty control element would fall out of the core and cause an unexpected increase in reactivity. It is true that the small reactivity worth of a single rod at Dresden would not be a serious hazard, but a similar failure in another reactor with only a few rods could cause a serious nuclear excursion.

Another reactor in which many difficulties with control rods were experienced is the SL-1 at the National Reactor Testing Station. A news release[18] following the accident on Jan. 3, 1961, reported approximately 40 cases of rods sticking in the last two months of operation. The situation was so bad that the rods had to be assisted manually during withdrawal, and, at shutdown after the last period of reactor operation, three rods had to be driven in by their motors. Two different factors were reported to have contributed to the difficulties. Friction in the seals through which the drive shaft penetrated the rack and pinion housing gave trouble in early operation, and compression of the control-rod shrouds by expanding boron strips probably caused the later difficulties. Whether these difficulties are in any way related to the accident that destroyed the reactor has not been determined.

On Aug. 27, 1960, the General Electric Test Reactor (GETR) was scrammed by an instrument failure in the primary coolant differential-pressure-monitoring system.[19] At this time the No. 2 control rod failed to scram and had to be inserted into the reactor by the manual "run-down" equipment. An inspection of the drive mechanism showed that the failure was in the scram latch device. The latch was manually released, and the rod fell. Subsequent inspection showed no visible defects, but lateral pressure applied by hand was found to produce frictional forces sufficient to prevent release. The scram-shaft guide bushings were lubricated with high-quality, low-oxidizing grease to correct this condition. The remaining five rods were inspected and lubricated, and tests of performance were made. Since no further difficulties were experienced, operation of the reactor was resumed. Bushing lubrication has been included in the preventive maintenance program, and a program for routine disassembly and overhaul of the drives, on a one-at-a-time basis, has been initiated.

This failure shows the importance of good rod-drive maintenance and testing procedures. Latch devices used for scramming must be highly reliable. Tolerances are usually small and adjustments critical. A sound design may be rendered ineffective by poor or incorrect maintenance procedures. Great care must be exercised to ensure that some faulty maintenance procedure does not cause all rods to fail for the same reason.

In addition to failures which prevent proper safety action, there are also failures in reactor safety devices which can initiate accidents or make trivial accidents much more severe. One of these was reported by AEC after the SL-1 accident, as follows: "Shield plugs, and other parts of the control-rod-drive mechanism, which had been placed in the reactor nozzles, were blown upward by steam or water pressure."[18] If the "other parts" mentioned included the poison section of one or more rods, this event surely increased the magnitude of the excursion. This type of failure in which poison can be blown from the core during an otherwise bearable accident or can fall from the core during operation or maintenance of the reactor is by far

more serious than simple failure of a rod to remove reactivity.

Most of the failures reported were the result of gradual changes in physical characteristics of the rods or drives. There have undoubtedly been many more unreported cases in which some slight malfunction has indicated a condition which could eventually cause failure. These conditions can be the result of poor design, faulty materials or fabrication techniques, improper maintenance methods, or other less evident factors. Constant attention to measurable performance characteristics can be quite effective in detecting system changes before they become serious hazards. Maintenance procedures for rods and drives must be as good as, if not better than, those used in instrumentation for reactor systems. Sound design is necessary, but continued effort is required to ensure that the design performance is retained.

(S. J. Ditto)

Analyses of Reactor System Transients Using Analog Computers

During the past decade, the general-purpose electronic analog computer has been employed more and more in the study of the transient behavior of physical systems. The analog computer lends itself well to the transient analysis of nuclear reactor systems, and it has been used quite extensively in this field.

In simulating a physical system on the analog computer, which implies the solution of the differential equations describing the behavior of the system, one is limited to *time* as an independent variable. This is an inherent characteristic of analog computers, since their integrators integrate with respect to time. Often other independent variables appear in the differential equations describing a physical system. Certain approximations must be made in order to simulate the systems on the analog computer. Various techniques are used in arriving at these approximations (difference equations to approximate differential equations, "cut-and-try" methods, etc.). The method employed is of secondary importance. The ultimate aim is to have the computer model behave in the same manner as the physical system under study.

With the aid of an analog computer, many of the hazards associated with the startup and operation of a new reactor system can be anticipated during the design stage. The causes of these hazards can be eliminated or provision can be made for the appropriate corrective action. It is also possible during the design stage to determine which parameters in the system require regulation or control and to evaluate possible systems for providing the required control. There are many approaches to the problem of arriving at a computer model of a physical system, and the final test of any model is the comparison of its behavior with that of the physical system being simulated.

A brief review of some documents dealing with analog computer investigations in the nuclear field is presented here. It is realized that there are many other publications dealing with this subject, and it is not the reviewer's intent to detract from these other publications.

An analog computer study of the 255-Mw Advanced Sodium Graphite Reactor (ASGR) system has been described in an Atomics International report.[20] The response of the neutron flux and various system temperatures to positive step changes in reactivity was studied. Several different magnitudes of reactivity change were used. The step changes in reactivity were initiated with the system in steady-state operation at full power, 50 per cent power, and 5 per cent power, respectively. The results of interest were as follows:

1. No coolant boiling in the fuel channels will occur as a result of inserting, in steps, up to 1 dollar of positive reactivity at full-power operation with no control-rod action.
2. The reactor coolant outlet temperature would be affected drastically if step changes in reactivity of 1 dollar or more were initiated with the system operating at 50 per cent power.
3. Coolant boiling will occur in the fuel channels as a result of a step increase in reactivity of 1 dollar or more initiated at steady-state 5 per cent power, if corrective action is not taken within 2 sec after initiation of the step increase.
4. The reactor power and temperatures will follow ramp increases in reactivity in a linear fashion.
5. Loss-of-coolant-flow accidents do not seem to present any problems. It appears that natural convection due to the coolant temperature gradient is sufficient to provide the necessary coolant circulation even with a total loss of pump power.

This appears to be a creditable investigation, and the results seem to be consistent with the expected behavior of the system.

Baker of the General Electric Aircraft Nuclear Propulsion Department has presented[21] a method for simulating reactor kinetic equations over many decades. In order to maintain an acceptable degree of accuracy in the analog computer, the voltage operating range is limited to two decades. During investigation of "startup" accidents or of transients in the "subpower" or very low power regions, the power may change by a factor of 10^{10}. The method used to get a multidecade range from a two-decade computer is briefly described as follows:

The machine voltage that is proportional to nuclear power is permitted to increase from 1 volt to 100 volts, at which point the voltage actuates a relay, which, in turn, switches the computer to the "hold" mode. In this mode the machine variables remain fixed at their attained values. The output voltages of all integrators are read and noted. These voltages are scaled down by a factor of 100, and the scaled values are used as initial conditions for these integrators in the next run. In the second run, the power advances from 1 volt to 100 volts, and again the relay is actuated. This procedure is repeated as many times as is necessary to get the required power. This method is straightforward and should achieve the objective. The reviewer points out one error in reference 21: in Fig. 2, page 5, amplifier 9 should be a summer rather than an integrator.

The simulation of a pump failure accident in the PWR has also been reported.[22] The approach used in this investigation was very good. In constructing the computer model, the performance of subsystems requiring approximations was compared with the expected performance obtained by theoretical methods or that expected from rational reasoning. These subsystem models were incorporated into the over-all system model only after their performance was judged to be satisfactory.

During the course of the computer investigation, the effect on the system of using different values for each of the parameters during a pump failure accident was determined in order to ascertain the parameters that appreciably affected the system behavior. The document[22] did not burden the reader with a maze of curves obtained by changing the unimportant parameters, as is so often the case. The authors of the report determined that the temperature coefficient of reactivity and the mean prompt-neutron lifetime were the parameters most influential in changing the transient behavior of the reactor system subsequent to a primary pump failure. A control concept to prevent adverse effects due to a pump failure was also proposed.

A transient analysis of an experiment in the liquid-metal-fueled reactor system has been conducted using an analog computer somewhat different from the usual general-purpose analog computer.[23] This computer, called Sampled-Data Simulator and Computer (SADSAC) essentially consists of a general-purpose analog computer in conjunction with a network of passive elements, electronic switches, stepping relays, etc. These less-expensive items perform some of the operations of operational amplifiers and multipliers, thereby decreasing the number of pieces of this expensive equipment required to simulate a given system. The author of the paper contends that the system, as simulated, would have required more than 1000 operational amplifiers if a conventional analog computer had been used; however, the simulation on his computer was accomplished with about 80 operational amplifiers, 11 stepping relays, 11 electronic switches, and the passive-element networks.

The frequency response of this circulating-fuel reactor was studied, and the results were compared to those obtained on a digital computer. The degree of agreement was very good. The system response to step changes in fuel flow rate, step changes in reactivity, and ramp changes in reactivity was investigated. These studies were conducted with various initial conditions of power, temperatures, etc. The results were those to be expected subsequent to the indicated perturbations. The economy in the number of operational amplifiers required to simulate the system appears to be a very significant accomplishment.

The analog-computer simulation of the Plutonium Recycle Test Reactor (PRTR) has also been reported in a well-written, comprehensive document.[24] In fact, the only critical comment the reviewer has is that the report is a bit lengthy. The approach used in deriving the computer model appears to be sound, and the results are consistent with those expected. The dynamic behavior of the system as a result of various perturbations was thoroughly investigated. This should give a very good idea of the

manner in which the actual physical reactor system will behave. (O. W. Burke)

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Section IV

PLANT SAFETY FEATURES

Radiochemical Plant Containment at Hanford

By R. E. TOMLINSON*

Radiochemical plants handle highly radioactive materials, such as liquids, gases, aerosols, and dusts, in very mobile states. Careful containment of these materials is necessary to prevent significant exposure of living organisms to the ionizing radiation from these materials.[1] The precautions taken to assure the required isolation of such radioactive materials make up the features that distinguish a radiochemical plant from other chemical plants. The protection of operating personnel from external gamma and neutron radiation is provided through the use of massive and hydrogenous shields,[2] respectively, and this feature will therefore not be discussed here. The status of containment technology is especially significant at present, however, because of recently reported incidents[3,4] and the current inclination on the part of private enterprise to enter the fuel-processing business.[5] This article discusses plant containment as it is currently applied to the radiochemical plants at Hanford.

The absolute containment of radioactive materials during chemical processing would be very expensive, if possible at all. The approach taken at Hanford has therefore been to hold the release of radioactive materials to the smallest practical rate. Environmental and ecological studies provide continuing information as to the kinds and quantities of radioactive material in the environment, permitting estimates of resulting human radiation exposure.

The technology of containing radiochemical "wastes," which has undergone continual improvement during the past 17 years of plant operation, was discussed in some detail in 1959 in the Congressional Hearings on Industrial Radioactive Waste Disposal.[6] More recently, radioactive waste treatment was discussed in *Nuclear Safety*.[7–9] The handling of plant effluents will therefore be discussed here only as it is directly associated with the problem of plant containment.

Chemical separations plants are of two general types: those which require massive shielding to protect operating personnel from penetrating radiation and those which do not. In the second category are the facilities for processing decontaminated uranium and plutonium. Regardless of shielding requirements, plutonium must be strictly confined to limit the introduction of this material into life cycles. The basic

*Roy E. Tomlinson is Manager of Advance Process Development in the Chemical Processing Department of the Hanford Atomic Products Operation. His association with the radiochemical processing business started in 1944 when he received training at the Oak Ridge pilot plant; he transferred to Hanford and assisted in the startup of one of the first radiochemical production plants, based on a bismuth phosphate precipitation process. In 1946 he joined a technical development group and during the next 10 years participated in the process development and design of the three major solvent-extraction plants at Hanford. (The Redox and Purex plants are still operating; the Metal Recovery Plant was deactivated after successfully recovering the uranium from the stored wastes from the original bismuth phosphate process.) For two of these years he was the supervisor of a pilot plant for developing the Redox and Purex processes using fully irradiated process materials. Since 1956 his primary function has been one of long-range planning and of defining goals for research and development. He is a member of the American Institute of Chemical Engineers and is working with a committee of the American Standards Association in developing standards for radioactive waste disposal.

containment techniques used in plutonium processing are the same as those used in the separations plants for processing irradiated fuels, and these two types of plants will be discussed together. The facilities used to process decontaminated uranium have no containment problems beyond those associated with natural-uranium processing, and these facilities will not be discussed.

General Plant Description

The chemical separations plants at Hanford employ solvent-extraction processes[10] to separate the uranium, plutonium, and fission products. The Purex plant employs tributyl phosphate (TBP) in a hydrocarbon diluent as the solvent and uses nitric acid as the salting agent. The Redox plant employs hexone (methyl isobutyl ketone) as the solvent and uses aluminum nitrate as the salting agent. Pulse columns and packed columns are used as contactors in the Purex and Redox plants, respectively.

Special railroad cars equipped with shielded casks are used to transport the irradiated fuel from the reactors to the remotely operated Redox and Purex plants. The irradiated fuel, carried in stainless-steel buckets, is dumped into dissolvers by a remotely operated crane. The aluminum jackets are dissolved in a sodium hydroxide–sodium nitrate solution, and the resulting solution is sent to underground storage tanks. The irradiated fuel is dissolved in nitric acid and processed by solvent extraction, as indicated above. Uranyl nitrate is sometimes passed through silica gel beds, if necessary, for additional decontamination from zirconium-niobium and is sent to a calcination plant for conversion to UO_3. Sometimes the Redox uranium product is also sparged with ozonized air to effect additional ruthenium decontamination prior to delivery to the calcination plant. Plutonium nitrate is purified in an anion-exchange facility at the Purex plant, and a similar treatment facility is being installed at the Redox plant. The purified plutonium nitrate is converted to metallic plutonium in the 234-5 Building. The fission-product wastes are concentrated as much as is practicable and are routed to underground storage tanks.

Since the fission products are highly radioactive, all operations, except the product purification and load-out steps, are conducted in a "canyon" behind massive concrete shields 4 to 6 ft thick. The processing equipment is contained in cells 12 to 14 ft wide and 27 to 150 ft long. The height of the Purex cells is 37 ft, whereas the height of the Redox cells varies from 24 to 30 ft; the Redox silo containing the solvent-extraction columns is 86 ft high. All cells (except the Redox silo) have removable stepped concrete cover blocks, and processing equipment is remotely installed or removed through these top openings. Remotely operated cranes traverse the canyon high above the deck formed by the cover blocks. These cranes are equipped with hooks, pipe grabbers, and impact wrenches to perform the necessary manipulations. Periscopes and closed-circuit television provide the necessary visual contact. All manipulations are controlled from shielded cabs on the cranes.

The canyon building shells are of reinforced concrete; the shells have no windows and a minimum of apertures for personnel and vehicle entry. The railroad access to each canyon is through a tunnel, with the railroad car being positioned below deck level. Doors are provided to isolate the railroad tunnel from the outside and from the canyon deck. All personnel entryways to the building are through air-lock vestibules to minimize problems of air balance and potential contamination spread.

The plutonium purification and load-out steps in Redox and Purex and all plutonium processing in the 234-5 Building are conducted in large glove boxes. The chemical processing operations are remotely controlled, but maintenance and many mechanical operations are effected directly through rubber gloves mounted in the glove-box faces. Equipment is usually moved into or removed from the glove boxes by use of the plastic-bag procedure,[11] in which the contaminated equipment or area is never permitted direct contact with the personnel work areas outside the glove box. Other transfer methods include the use of sphincter seals, air locks, and transfer cans to minimize contact between operating personnel and contaminated materials. The 234-5 Building shell is of insulated sheet metal, and all joints are sealed. There are no windows, and access is through air locks.

Containment Zones

The plant facilities are designed to confine radioactive material within the process equipment to the greatest practical extent and to

provide multiple barriers to its escape. It is convenient to think of each plant as having four zones representing different degrees of contamination.

Zone 1 includes those areas which are not intended to be contaminated at any time, such as control rooms, offices, chemical makeup areas, and ventilation intake equipment. These areas are checked routinely for contamination, and contaminated areas are cleaned promptly if found. Airborne contamination is measured every shift. Personnel move about freely in these areas. Personnel monitoring is required before eating and before leaving the building.

Zone 2 includes those work areas which are not normally contaminated but which may become contaminated through abnormal processing conditions. Examples of areas in this zone are the galleries provided on the cold side of the shielding wall for process and utility influent control and for sampling process solutions, the work areas outside the glove boxes, and the laboratories. From a containment viewpoint, the canyon area above deck level may be considered zone 2 when the cell covers are in place. Since the canyon deck is never completely free of contamination, this area is rarely entered; personnel do enter the canyon when entering or leaving the control cab of the remotely operated crane, but they are about 30 ft above the canyon deck and are protected by a parapet shield. The areas behind the operating control panels are considered either zone 1 or zone 2, depending on the type of connection between the instruments and the processing equipment. Where pneumatic lines connect the instruments directly with process equipment, the area is considered zone 2; where only electrical leads are used, the area is considered zone 1. Personnel usually enter a zone 2 area through a changeroom, where they change to protective clothing. The protective clothing is discarded in the changeroom when the person leaves the zone 2 area. In those cases where the zone 2 area is physically removed from a changeroom, the protective clothing worn into the area is discarded at the zone 2 exit when the person leaves the area. Personnel are carefully checked for contamination when they leave a zone 2 area. Most working areas subject to contamination are monitored by permanently installed samplers and instruments. Such areas are also routinely monitored with portable instruments while work is in progress, and all zone 2 areas are checked at regular intervals. When contamination is found, it is removed to the extent practical, and the residual contamination is stabilized with paint.

Zone 3 includes the cells or glove boxes which contain the process equipment; laboratory hoods and the enclosures housing sampling equipment in the sample galleries are also included. These areas are heavily contaminated but are kept nominally free of macroscopic quantities of process materials. Materials that leak from the process equipment are flushed into sumps and returned to the processing complex or sent to underground storage. It is not intended that personnel enter these areas, and months of extensive decontamination processing would be required to permit such access.

Zone 4 consists of the process equipment itself, except for the remote-control equipment. The process equipment is nominally airtight, except for the dissolvers where the irradiated fuel is introduced; a significant quantity of air in-leakage is required here for process reasons. Leaks of process solutions or utilities are tolerated to the extent that they do not interfere with the process, corrode the equipment dunnage or cells, or cause the loss of significant quantities of product.

Ventilation

In the Redox and Purex buildings, the zone 1 areas and the zone 2 areas outside the canyons are supplied with washed, filtered, and heated air that is exhausted to the atmosphere. The zone 2 area exhaust air is filtered before release. Each canyon is supplied with washed, filtered, and heated air through fans located near the roof level. The air flows over the parapet wall, which partially shields the crane control cab, and flows down to the deck level. The air then enters the process cells through cracks (approximately $\frac{1}{2}$ in. wide) between the cover blocks. The rate of air flow through the cells is adjusted to maintain the cell temperature at a tolerable level, i.e., approximately 50°C. The air then enters an exhaust plenum adjacent to the cells through ports in the cell wall. The exhaust air is filtered, as discussed below, and is exhausted to the atmosphere through a 200-ft stack.

The ventilation pattern of the 234-5 Building is comparable to that of the Redox and Purex buildings. Because of the physical layout, the

zone 2 areas consist of many rooms containing glove boxes and laboratory hoods. For these rooms the ventilation air is supplied and exhausted through plenums and is filtered before being exhausted through a 200-ft stack to the atmosphere.

A pressure differential is maintained between the different zones, with the processing equipment generally having the lowest absolute pressure; the flow of air is thus maintained from the less radioactive (zone 1) to the more radioactive (zone 4) areas. A notable exception to this rule is some of the Redox processing equipment; the inert-gas blanketed system is maintained at a pressure slightly positive with respect to the cell to preclude the accumulation of an explosive air-hexone mixture in the processing equipment. Typical pressures and rates of air change maintained in the operating buildings are listed in Tables IV-1 and IV-2, respectively.

Table IV-1 TYPICAL PRESSURES IN PLANT AREAS

	Pressure (in. H_2O) relative to barometric pressure		
	Purex	Redox	234-5 Building
Zone 1	+0.2	+0.2	+0.25
Zone 2:			
Work areas	−0.1	−0.1	+0.13
Canyon deck	−0.3	−0.2	
Zone 3:			
Cells or hoods	−0.8	−0.25	−0.8
Exhaust plenum	−1.4	−0.6	−2 to −4
Zone 4	−3 to −30	−0.2 to −30	−2 to −4
Ventilation exhaust (downstream of filter, zone 2):			
Fan intake	−15.0*	−9.0	−6.5
Base of stack	+1.5	+1.0	+0.1

*A damper provides about 9 in. of pressure drop and may be opened to compensate for filter plugging. This action has not been required.

Table IV-2 TYPICAL AIR FLOW IN PLANT AREAS

	Air changes per hour		
	Purex	Redox	234-5 Building
Zone 1	8	8	8
Zone 2:			
Work areas	8 to 20	3 to 8	12
Canyon deck	2	2	
Zone 3	2 to 12	8	*

*The glove boxes are serviced with recirculating dry air; the net flow of air through the hoods is very small.

Ventilation Exhaust Filters

Four types of filters are in use to remove radioactive particles from plant gaseous effluents. These filters are composed of sand, glass fibers,[12] fire-resistant glass and/or asbestos paper,[13] and the cellulose-asbestos paper filter developed for the Chemical Warfare Service.[14] The cellulose-paper filter is being used in only a few places; the fire-resistant type is preferred and is used in all the exhausts from the zone 2 areas. The glass-paper filters are in unitized, fire-resistant frames, and are replaced whenever the pressure drop becomes excessive or the filtering efficiency declines. The initial resistance to air flow is about 0.9 in. H_2O at rated capacity; the individual units are rated at 25 to 1000 cfm. The Redox and Purex canyon ventilation filters are permanently installed, and a new filter would be built if the operating performance fell below requirements and could not be restored by external manipulation.

The Redox ventilation filter consists of layers of sand and gravel in a concrete shell equipped with air distribution tiles. There are seven layers of materials, decreasing in particle size from coarse gravel on the bottom (particles 2 to $3\frac{1}{2}$ in. in diameter) to a layer of fine particles (20 to 40 mesh); air flow is upward through the unit. The filter removes about 99.7 per cent of the particles having a diameter of 1 μ. The unit operates with a pressure drop of 8.3 in. H_2O at a face velocity of 5 ft/min.

The Purex ventilation filter[15] is composed of glass fibers especially tailored for strength and moisture resistance. The prefilter is composed of fibers having a diameter of 30 μ. This section is dump-filled to a depth of 7 ft, and and the fibers have a bulk density varying between 1.5 and 6 lb/cu ft. The air flows down through the bed at the rate of 50 ft/min. The fine filter is composed of glass fibers having diameters of 1.3 to 2.5 μ and a bulk density of about 1.2 lb/cu ft. This section, which is composed of 1-in.-thick mats held in frames, operates at a face velocity of approximately 20 ft/min. The edges and junctures are carefully sealed to avoid bypassing. The unit retains 99.99 per cent of the particles that are approximately 1 μ in diameter, and it operates with a pressure drop of approximately 4 in. H_2O.

Process Gaseous Effluents

In the 234-5 Building, air exhausted from the zone 3 and 4 areas is filtered through paper filters at least twice, and in most cases three times, before release with the ventilation air. The fire-resistant type of filter is used in all locations except where process off-gases would attack the glass fibers. In the Redox and Purex plants, process off-gases are handled in three systems: the dissolver off-gas, the vessel vent, and the condenser vent systems.

The dissolver off-gas is cooled and partially stripped of nitrogen oxides in a downdraft condensing tower. The cooled gas is then heated to about 200°C and passed through a silver reactor.[15] This reactor is composed of porous ceramic packing impregnated with silver nitrate. It normally removes more than 99.5 per cent of the radioactive iodine entering the unit. The gas is then passed through a packaged filter that typically consists of multiple layers of 30-μ-diameter moisture-resistant glass fibers packed to different densities. The unit is contained in a stainless-steel canister and is built up of a 10-in. layer of glass fibers at a density of 1.5 lb/cu ft, a 6-in. layer with a density of 3 lb/cu ft, a 12-in. layer with a density of 6 lb/cu ft, and a 12-in. layer with a density of 9 lb/cu ft. In a separate building the gases are passed through a wet scrubber to remove more nitrogen oxides and radioiodine and are exhausted by a steam or air jet to the ventilation exhaust air as it enters the stack. A second packaged filter is being installed ahead of the wet scrubber to provide additional assurance that radioactive particles will be contained under any foreseeable condition.

The vessel and condenser-vent effluents are passed through packaged filters of glass fibers of the same type as is used in the dissolver system. The filtered off-gases are discharged to the ventilation exhaust plenum by steam jets. The Purex vessel vent system also contains a silver reactor to retain radioiodine. The effluent from the Redox ozonization step is discharged into the ventilation exhaust plenum without prior treatment.

Abnormal Conditions

The integrity of the containment system must be protected under emergency conditions to limit contamination of the environment. In the event of a power failure, the flow of ventilation air is automatically maintained by separate exhaust fans powered by steam turbines. These fans maintain the zone 3 areas at negative pressures with respect to the atmosphere. In addition, a few of the fans that maintain air flow to and from zones 1 and 2 are also kept running by electricity supplied by a steam-turbine-driven emergency generator at the central power house.

In the event of a sudden energy release in the process equipment, such as by a rapid chemical reaction, the energy must be dissipated without destroying the integrity of the containment. The processing equipment may be deformed or ruptured in the incident, and the processing area may be grossly contaminated. In Redox and Purex, the barrier walls of the process cells are constructed of concrete and have sufficient mechanical strength to contain any flying debris. Any sudden increase in gas volume is bled off through the normal process and ventilation exhaust systems. The ventilation filters are protected from any shock wave by physical isolation; the ducts connecting the processing areas with the filters are long and change direction several times. Each processing area is connected with the exhaust plenum through a limited area port so that any increase in pressure at the filter is relatively gradual and limited in magnitude. In some cases the pressure surge may blow radioactive materials back to a zone 2 area; for example, some radioactive solution may be blown out of a sampler, a rubber glove may be ruptured in the face of a glove box, or a panel might be blown out of a hood. The quantity of process material transported to a zone 2 area is generally small, however, and the normal containment capability of the zone 2 area is sufficient to contain the material.

Containment Performance

The ability of the separations plants to contain radioactive materials under normal operating conditions was reviewed in some detail in the Congressional Hearings on Industrial Radioactive Waste Disposal.[6] Over the years, successive experiences with the emission of contaminated particles, radioiodine, and radioactive ruthenium led to the development of various types of filters, the silver reactors and backup scrubbers, and Redox process changes, respectively. The ventilation and process effluent gases are sampled continuously at the stack and are analyzed for radioactivity content. The

indicated average daily release of radioactivity from the three plant stacks during 1960 was 1.0 curie of I^{131}, 0.01 curie of Zr^{95}-Nb^{95}, 0.007 curie of Ru^{103}, 0.006 curie of Ru^{106}, 0.001 curie of Ce^{141}, and 0.00003 curie of total alpha (presumed to be plutonium).[16] Many of the analytical results are at or near the limit of detection, and therefore the actual releases may well be lower than indicated. These values are typical of plant performance and vary little from year to year. The I^{131} released may be compared with the self-imposed working limit[17] of 10 curies/week. This limit is based on studies by personnel of Hanford Laboratories showing that sheep would suffer no detectable damage if grazed continuously at the plant perimeter under these conditions.

The containment efficiency of the multibarrier design has also been demonstrated under abnormal conditions. Power failures have occurred repeatedly, but no contamination release has resulted. In addition, several incidents that occurred have involved rapid release of energy of chemical or mechanical origin. In no case has radioactive material been transported to a zone 1 area or to the environs as a direct result of such energy release. Two of the more energetic incidents are described briefly below.

In February 1958 a silver reactor in the Purex plant erupted.[18] The bottom of the reactor was ripped open, the reactor and dunnage were elevated a few feet, and the restraining piping and dunnage were bent severely. A large quantity of a fine brown powder carrying radioiodine was deposited over all the equipment in the cell. No new contamination was detected outside the cell, exhaust plenum, and filters.

In April 1960, irradiated uranium caught fire and burned in a Redox dissolver, as described in the June 1961 issue of *Nuclear Safety*.[19] The fire occurred because bare uranium was left uncovered for two days. Holes were burned in the dissolver cooling coil and vessel wall. When water contacted the hot metal, the system was pressurized, and a large quantity of radioactive dust was released to the cell, vent, and filters. No appreciable increase was noted in the rate of fission-product release to the atmosphere.

Although the multiple-barrier concept has proved to be a sound and reliable basis for the containment of radioactive materials in Hanford separations plants, this concept is not a panacea. Considerable foresight, continuous vigilance, and strict adherence to established operating procedures are required to contain the radioactive materials in the intended areas. Improved containment concepts are being sought, to decrease the amounts of radioactivity released and to reduce the total cost of containing the radioactive materials. (R. E. Tomlinson)

Hazards Analysis of Chemical Processing Facilities

The hazards associated with the operation of a chemical processing plant may be as great or greater than those associated with a reactor, as measured either in terms of the total fission-product inventory or the activity released in a nuclear excursion. On the other hand, many radiochemical operations are performed in which the quantities of fissionable materials involved are so small that accidental criticality is out of the question, but in which the amounts of activity that could be released by potential chemical reactions or mechanical failures are so great that special containment provisions are required. Containment provisions for such operations generally consist of a ventilated cell in which the remotely operated equipment is located and the ventilated building in which the cell is located. The ambient pressure in the ventilated building is maintained below atmospheric pressure, and the pressure in the cell is maintained below that in the building.

An attempt to evaluate the personnel hazard following a maximum credible accident in such a chemical processing or fuel-handling facility is described in this review. Calculational methods similar to those employed for reactor hazards evaluation studies are used to estimate the amount of activity released and its dispersion in the atmosphere. Two types of accident are considered: (1) a chemical explosion with release of radioactivity as a suspended aerosol or dust and (2) a criticality incident followed by release of the gaseous fission products, as well as release of the initially present radioactivity as an aerosol or dust. In cases where such an accident is considered credible, the criticality incident is by far the more serious.

If a processing or fuel-handling facility accident is minor, i.e., pressurization of the cell does not occur, the release of radioactivity is confined to the vessel or cell off-gas system, with subsequent release of the gas to the environment through scrubbers, filters, and off-

gas stacks. However, if an accident causes pressurization of the cell, activity can escape into the operating area within the building and even contaminate the ground or nearby buildings. Protection of personnel and the surrounding area or buildings depends on such factors as cell tightness, reliability of the cell ventilation system, building tightness, and reliability of the emergency ventilation system of the building.

The hazard of a given processing or handling operation to the building occupants if the cell is pressurized has been determined analytically and has been found to be directly proportional to the product of the following factors: (1) concentration of activity in fuel solution or other medium, (2) total quantity suspended per unit volume of air, (3) effective cell leakage rate constant, (4) square root of the average pressure difference, and (5) duration of cell leakage. If the hazard is evaluated for the area outside the building, the following additional factors are involved: (1) the effective building leakage rate, (2) the duration of building leakage, and (3) a variable which expresses an effective dilution factor calculated from meteorological conditions.[20]

Equations for Maximum Downwind Dose

Equations have been developed by Unger and Arnold[21] and modified by Arnold et al.[22] to express the maximum downwind dose resulting from the release of aerosols produced from radioactive solutions and from gaseous (volatile as well as noble gas) fission products generated by a criticality incident which produced 10^{18} fissions. The terms used in these equations are defined in the appendix to this review.

If the MPC_M for the mixture of fission products in the solution is known, the exposures may be determined in a single calculation. If this MPC_M value is not known, the dose must be summed for the individual isotopes.

These dose equations have been used in the evaluation of the containment of a number of ORNL chemical processing facilities.[23] (Reference 23 is representative of and refers to several hazards reports on ORNL chemical processing facilities.) The doses have been evaluated for several different types of release, and four of these are as follows:

1. Activity release through vessel off-gas system (in rem). The downwind dose for an aerosol is

$$D_v = \frac{1.72 \times 10^{-6} A_v m k C_F}{(\mathrm{MPC})_M}$$

The downwind dose for gaseous fission products from 10^{18} fissions is

$$D_{v,i} = \frac{0.27\, \lambda_i y_i k e^{-\lambda_i t_D}}{X_i (\mathrm{MPC})_i}$$

2. Activity release through cell off-gas system (in rem). The downwind dose for an aerosol is

$$D_c = \frac{2.86 \times 10^{-8} V_c m k C_F}{(\mathrm{MPC})_M}$$

The downwind dose for gaseous fission products from 10^{18} fissions is

$$D_{c,i} = \frac{0.27\, R \lambda_i y_i k e^{-\lambda_i t_D}}{(R + \lambda_i) X_i (\mathrm{MPC})_i}$$

3. Activity release to the building (in rem). The dose following a 2-min inhalation for an aerosol is

$$D_s = \frac{1.67 \times 10^{-7} m V_c K_c t_E}{V_B (\mathrm{MPC})_M} \times \left(\frac{101.5\, V_E}{V_c} - 0.25\, P_I\right)^{0.5}$$

The dose following a 2-min inhalation for gaseous fission products from 10^{18} fissions is

$$D_{s,i} = \frac{0.157\, K_c \lambda_i y_i t_E}{V_B (\mathrm{MPC})_i} \left(\frac{1 - e^{-\lambda_i t_E}}{2}\right) \times \left(\frac{101.5\, V_E}{V_c} - 0.25\, P_I\right)^{0.5}$$

4. Activity release from the building. Activity may be released from various points in the building if the building is pressurized with respect to the atmosphere and may be released through the building emergency-ventilation exhaust-system stack. The following equations express the sum of the maximum exposure (in rem) due to activity released from the building and from the stack. These maxima do not necessarily occur at the same distance from the building. If the building leakage dose is controlling, it is necessary to use a stack-dilution factor, which would have to be calculated separately for that distance at which the building

exposure is a maximum. The downwind dose for an aerosol is

$$D_D = \frac{4.77\times10^{-9}\,K_c m V_c}{(\mathrm{MPC})_M}\left(\frac{101.5\,V_E}{V_c} - 0.25\,P_I\right)^{0.5} \times \left(k_L L_s + \frac{C_F}{10}k_B\right)$$

The downwind dose for gaseous fission products from 10^{18} fissions is

$$D_{D,i} = \frac{4.51\times10^{-3}\lambda_i y_i K_c}{(\mathrm{MPC})_i}\left(\frac{101.5\,V_E}{V_c} - 0.25\,P_I\right)^{0.5} \times \left(\frac{\frac{K_B}{60}k_L\left\{1-e^{-[(K_B/60)+\lambda_i]t_L}\right\}}{\frac{K_B}{60}+\lambda_i} + \frac{R_B k_B e^{-\lambda_i t_D}}{R_B+\lambda_i}\right)$$

Fallout of Radioactive Particulates

The second important hazard, i.e., in addition to the total internal or external dose received by operating personnel or personnel downwind of an accident, could arise from deposition of radioactive particles. If by deposition of airborne particles the surrounding terrain becomes contaminated beyond permissible levels for alpha or beta-gamma activity, it may be necessary to decontaminate large areas surrounding the affected facility. Fallout is serious only for the fraction of the activity that escapes through a building leak, since any reasonable release from a tall stack will be diluted by many orders of magnitude before reaching ground level.

The deposition of particles results from the release of an aerosol following either a chemical or a nuclear explosion. The total deposition of particulate matter at any point x,y is given by the expression:[24]

$$\omega = \frac{2QVg}{e\pi C_y C_z \bar{u} x^{2-\bar{n}}}\exp\left(-Z_1^2 + y^2/C_y^2 x^{2-n}\right) \times \exp\left(-4Vg\; x^{n/2}/n\bar{u}\pi^{1/2}C_z\right)$$

where

$$Z_1 = h - \frac{xVg}{\bar{u}}$$

Leaktightness Specifications for Building Structure

Definite criteria for cell and building leaktightness or ventilation-system operating characteristics cannot be set because the true values depend on the type of operation and the specific activity and type of radioactive material involved. The variation of downwind dose is linearly proportional to the cell leakage rate but somewhat less than linearly proportional to the building leakage rate.

Investigations of the effect of cell and building leakage rates indicate that a tight cell is much more important than a tight secondary container, i.e., the building. Since the leakage of activity from the secondary container following leakage from the cell is directly proportional to the concentration of the activity, it follows that the downwind dose is also directly proportional to the total leakage from the cell. On the other hand, the material leaks from the building along two paths: (1) through the building leak and (2) through the filters (since even absolute filters are not 100 per cent effective) of the secondary container ventilation system. If the secondary container ventilation system exhausts at a low elevation, such as the roof or a short stack, the dose due to the inefficiency of this ventilation system will be much greater than that resulting from 20-sec-duration leakage from the building. A change of a factor of 10 in building tightness might change the total downwind dose only 10 per cent. However, if the exhaust-system dilution factor can be decreased by a factor of 100, such as by exhausting through a 250-ft stack, a much smaller dose will be received as a result of filter inefficiencies. In this case a factor of 10 increase in building leakage rate, L, will increase the dose downwind, D, by a factor of 5. An illustration of these effects is given in Table IV-3.

Table IV-3 EFFECT OF BUILDING LEAKAGE RATE ON DOWNWIND DOSE

Building leakage rate	Downwind dose for a stack dilution factor of 2×10^{-3}: From building leakage	2×10^{-3}: From stack	2×10^{-5}: From building leakage	2×10^{-5}: From stack
L	$2D$	$200D$	$2D$	$2D$
$10L$	$20D$	$200D$	$20D$	$2D$

The results of these analyses indicate that, wherever possible, the cell should be tightened as much as possible and that the building ventilation system should have a low stack-dilution factor. The main purpose of a tight secondary container is to provide a second line of defense that would help to contain the activity from a major cell leak resulting from a catastrophic accident or operational error.

Effects of Chemical and Nuclear Explosions

In most facilities where explosions would constitute the maximum credible accident, the magnitudes and effects of chemical and nuclear explosions would be similar because of the similarity of the process materials and equipment. If it were possible to convert approximately 3 lb of organic material to the nitrate in a chemical processing facility and if this material were assumed to be equivalent to TNT in explosive violence, the explosion of the nitrate material could release almost instantaneously 5700 Btu of energy and ~100 cu ft of steam and gaseous combustion products. The shock wave from such an explosion would certainly rupture and scatter the contents of a vessel and would cause a pressure of ~800 lb/sq ft at a distance of 15 ft from the center of the explosion. The explosion of a hydrogen-air mixture in the vapor space of a process vessel would also rupture the vessel and generate ~100 cu ft of gas and steam, but less powerful shock waves would be generated than in the explosion of nitrated organic material.

In the facilities in which a nuclear accident is credible, the maximum nuclear accident is considered to be due to a single burst of 10^{18} fissions that is terminated by the rupture of the process vessel.[25] An initial burst of 10^{18} fissions is considered a conservative limit for solutions of fissile material in vessels of the size used in most processing facilities, since the initial and the most powerful bursts that have been observed in approximately 15 prompt critical excursions that have occurred in critical-assembly laboratories and production plants have been of the order of 10^{17} fissions or less. In solutions and water-moderated lattices, the energy released in 10^{17} fissions is apparently sufficient to shut down the nuclear reaction temporarily as a result of thermal expansion and radiolytic gas-bubble formation. If a shutdown device is not actuated following the initial surge, the reaction may tend to recur and gradually boil away the moderator or expel fissile material from the reaction vessel until permanent subcriticality is obtained. This type of accident is typified by a recent Idaho Chemical Processing Plant incident[26] in which approximately 4×10^{19} fissions occurred over a period of approximately 7 min before permanent subcriticality was achieved. A recurring reaction of this type when contained in a process vessel would, however, have less serious consequences than that due to a burst of 10^{18} fissions accompanied by rupture of the process vessel.

Experience and calculations indicate that it is not incredible that a vessel would rupture in a nuclear accident. Slight destructive effects were manifested in a prompt critical excursion in 1956 at the ORNL Critical Experiments Laboratory.[27] In this excursion, 1.6×10^{17} fissions in a single burst occurred in a solution of about 60 liters of fully enriched UO_2F_2 in a 30-in.-diameter open-top cylinder, and the bottom of the cylinder, which was made of 0.5-in.-thick 2S aluminum, was noticeably distorted by a downward force. Calculations indicate that in reprocessing solutions it may be possible to assemble reactivity at maximum rates of the order of a few dollars per second. The peak pressures which might occur, depending on the inertia of the liquid in the vessel and the characteristics of the venting system, have been calculated to be of the order of 10 to 100 atm or greater. Such pressures would be sufficient to rupture most process vessels.

The effects of the assumed nuclear burst of 10^{18} fissions would be similar to the effects of the maximum credible chemical explosion in that the process solution would be scattered throughout the process cell. It is probable, however, that the nuclear burst would consist of a low-order explosion with a negligible shock wave. Approximately 30,000 Btu of energy and 2.0 scf of gas would be released in the burst. It is incredible that more than 100 cu ft of steam could be formed, since a larger steam volume would constitute an extremely effective shutdown device and would probably prohibit a burst of 10^{18} fissions. The effects of a nuclear accident in addition to the effects of a chemical explosion would be that personnel would be subjected to prompt neutron and gamma radiation, and "fresh" gaseous fission products would be released to the ventilation systems.

Particulate Material in Ventilation Streams

The properties of aerosols in air streams provide a reasonable basis for estimating the quantity of activity that would escape from primary containment zones in the event of an accident, since air streams, unless they are exceedingly turbulent, can contain only a small fraction of solid or liquid suspended matter. Radioactive-solution particles would be suspended in cell and vessel air as a result of the maximum credible accident. These liquid particles, which would vary in size from a fraction of a micron to several hundred microns, depending on the velocity of the air stream, would have essentially the same composition as the solution from which they have escaped. For superficial velocities less than about 0.15 ft/sec, an aerosol formed by vigorous mixing of a solution with air[28,29] is metastable, it contains particles with a mean particle size of approximately 10 μ, and it has a concentration of the order of 10 to 15 mg/m^3. After several changes of direction of flow in the off-gas line, the mean particle size of the metastable aerosol will be of the order of 3 μ, and the concentration will be about 10 mg/m^3. At this point the aerosol will contain equal weight fractions of particles in the size ranges of $<0.4\,\mu$, 0.4 to 1.3 μ, 1.3 to 3 μ, 3 to 5 μ, and $>5\,\mu$. It has been shown[30] that the concentration, as well as the particle-size distribution, of particles smaller than about 20 μ in gas is relatively insensitive to the total quantity of particulate matter in the gas.

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Appendix: Nomenclature

A_v = vessel off-gas flow rate at filters, cfm

A_c = cell ventilation rate, cfm

C = atmospheric diffusion constant where subscripts y and z refer to y and z directions

C_F = concentration of activity in gas stream leaving filters, mg/m^3

D_c = downwind dose from aerosol released through cell off-gas system, rem

$D_{c,i}$ = downwind dose from fission products released through cell off-gas system, rem

D_D = downwind dose from aerosol released from building (leakage + ventilation air), rem

$D_{D,i}$ = downwind dose from fission product i released from building (leakage + ventilation air), rem

D_S = dose to building personnel from aerosol, rem

$D_{S,i}$ = dose to building personnel from fission product i, rem

D_v = downwind dose from aerosol released through vessel off-gas system, rem

$D_{v,i}$ = downwind dose from fission product i released through vessel off-gas system, rem

h = height of release, m

k = stack dilution factor, sec/m^3

k_B = building ventilation system stack-dilution factor, sec/m^3

K_B = building leak constant, building volumes/min at 0.3 in. H_2O

K_c = cell leakage rate constant, cell volumes/min at a positive pressure of 2 in. H_2O

$k_L = 2/\pi e \bar{u} h^2$ = leak dilution factor, sec/m^3

$L_S = 1 - e^{-K_B t_L/60}$

m = concentration of activity in solution, curies/mg

$(MPC)_i$ = maximum permissible concentration of fission product FP_i in air (40-hr basis)

$(MPC)_M$ = maximum permissible concentration in air (40-hr basis) for mixed fission products in solution

n = atmospheric stability parameter

P_I = initial cell vacuum, in. H_2O

Q = total quantity released, mg (or curies)

R = cell removal constant ($= A_c/60\,V_c$), cell volumes/sec

R_B = building venilation rate, building volumes/sec

t_D = decay time between fission and stack discharge, sec

t_E = evacuation time for building, sec

t_L = time building leaks, sec

$\bar{u}$ = average wind velocity, m/sec

V_B = building volume, cu ft

V_c = cell volume, cu ft

V_E = volume of gas generated by explosion, cu ft

V_g = particle settling velocity, m/sec

x = distance downwind, m

X_i = decontamination factor for fission product FP_i

y = distance crosswind, m

y_i = fission yield for FP_i

λ_i = decay constant for fission product FP_i, sec^{-1}

ω = total deposition due to fallout, mg/m^2 (or curies/m^2)

Contamination of Fuel-Element Surfaces

An important design criterion for most reactors utilizing solid fuel is that the fissionable material contamination of the coolant should not be excessive. Even small amounts of fission-product activity in the coolant could lead to test loop or plant operating problems, particularly with respect to maintenance of the coolant system. In addition, the radiation and the ingestion hazards associated with the gaseous fission products that might leak out of the coolant system could limit access time in areas with recirculating air supplies. This latter problem is of special importance in nuclear plants for submersible vessels that operate under recirculating ventilation conditions when submerged. It may also be significant in other power reactors or, possibly, research reactors, depending upon specific design.

Solid fissionable material is usually clad with a metal or an alloy that is chemically compatible with the fuel and corrosion resistant to the coolant to prevent the release of fission-product activity to the reactor coolant. Rigid quality control of the fabrication process and thorough nondestructive testing are enforced to ensure that the fuel is hermetically sealed within its protective sheath. In many cases the cost of inspection represents a large fraction of the total cost of the fabricated part. With all attention directed to this problem, it is not surprising that a simple phenomenon such as inadvertent fissionable contamination of fuel-element surfaces during fabrication might be overlooked. In such an event, it is conceivable that all the effort spent in carefully jacketing the fuel may be negated because the surface contaminant will release fission products to the coolant and the effect will be the same as the effect of a defect in the cladding. Although fissionable contaminants could be present on any of the core or test loop components, the fuel elements have the greatest potential for contamination because the fabrication sequences involve handling of the fissionable fuel.

Investigations at Knolls Atomic Power Laboratory (KAPL) and ORNL have revealed the presence of U^{235} on the surfaces of Zircaloy-2-clad and aluminum-clad fuel elements, respectively, in excess of the amount of U^{235} in the cladding material prior to fabrication of the fuel element. In the KAPL studies,[31] the contamination found on the surfaces of Zircaloy-2 fuel elements from industrial fabricators was in excess of 10^{-7} g of U^{235} per square decimeter, which is considerably more than the 5 ppm uranium impurity normally present in Zircaloy-2 stock material, considering any reasonable thickness of Zircaloy. In studying the problem, it was found that autoclave pressure vessels used to conduct corrosion tests of fabricated fuel plates had a fixed contamination level, measured by alpha-particle counting, as high as 19,500 dis/(min)(dm^2), and it was demonstrated that the high-temperature water had the ability to transfer U^{235} contamination from the autoclave walls to the Zircaloy-2 surfaces. The possibility of diffusion of uranium from the fuel alloy to the surface of the Zircaloy-2 was also studied, but diffusion was found to be negligible even after heat-treatment of fuel plates at 730°C. In examining the fabrication process, it was found that, although surface and subsurface contamination of fuel plates occurred during early mechanical working and heat-treating operations, the contamination was removed by subsequent pickling. It was concluded[31] that "the contamination observed on completed assemblies was introduced and fixed in the final corrosion testing."

The hazards created by operating a contaminated pressurized-water reactor system in an area with restricted ventilation were also investigated.[31] It was found that, with a total quantity of U^{235} from surface contamination as low as 1 to 10 mg of U^{235} per core, an airborne radiation hazard might result from primary coolant leakage. It was concluded, however, that the exposure that would be received by personnel from shutdown radiation, coolant piping, and system components during maintenance work would not become excessive. For the calculations it was assumed that (1) a primary coolant leak could result in fission-product ejection into the plant's atmosphere; (2) the total holdup time between leakage into the secondary system and ejection into the plant's air was small compared with the half lives of the isotopes considered; and (3) 75 per

cent of the external dose would result from 5.27-day Xe^{133}, which emits a fairly soft beta ray, and about 90 per cent of the internal dose would be caused by 18-min Rb^{88}, which accumulates in the muscle. At equilibrium the Xe^{133} and Rb^{88} air activities would be 1.5×10^{-5} $\mu c/cm^3$ and 1.3×10^{-7} $\mu c/cm^3$, respectively. The calculations revealed that, after 60 days of confined operation, the total external dose from Xe^{133} would be 1.73 rem, and the internal dose from Rb^{88} would be 0.005 rem. It was assumed that the spot coolers would not remove any of the particulate fission products and that 54-day Sr^{89} would present the worst internal radiation hazard. Thus an internal dose of about 0.74 rem would occur. These doses are excessive; however, it was pointed out[31] that an air ventilation rate of only 30 cfm would reduce the air activity so that the external and internal dosages would meet the AEC recommendations with or without spot coolers.

Research reactors of the pool type, such as the Bulk Shielding Reactor (BSR), or of the tank type, such as the Low-Intensity Test Reactor (LITR) and the Oak Ridge Research Reactor (ORR), are somewhat more accessible than pressurized-water systems, and the problems of maintenance in radiation fields after long-term operation may become significant. An investigation of the LITR coolant by Moeller and Leddicotte[32] revealed an I^{133} activity of 8×10^{-4} $\mu c/ml$, which could be traced to a total surface contamination of 5.5 mg of U^{235} on the 26 fuel elements comprising the LITR core. Analyses of scrapings from the contaminated aluminum cladding indicated that the U^{235} contamination, if uniform throughout the metal scrapings, was about 20 ppm, or 1.24×10^{-6} g/dm^2. This may be compared with the 0.5×10^{-9} g of U^{235} per square decimeter, which would be the minimum expected surface contamination based on the 0.008 ppm U^{235} normally present in wrought aluminum stock.[33]

Studies have also been conducted at ORNL on the effect of the fabrication process on U^{235} surface contamination,[33,34] as well as on diffusion of uranium from the uranium-aluminum alloy fuel matrix to the cladding surface. The data given in Table IV-4 show conclusively that U^{235} can be transferred from contaminated rolls in rolling mills to the aluminum plate surface; however, most of the contamination can be removed by pickling. The data also indicate that

Table IV-4 CONCENTRATION OF U^{235} IN 3-MIL SCRAPINGS FROM THE SURFACE OF ALUMINUM-CLAD URANIUM-ALUMINUM ALLOY FUEL PLATES AFTER VARIOUS STEPS IN PLATE FABRICATION

	U^{235} concentration (ppm) in fuel plates	
	From hot-rolling mill contaminated with U^{235}	From partially contaminated hot-rolling mill
After hot rolling at 600°C	180.0	8.9
After flux annealing at 600°C for 40 min, etching in 2% HF–15% HNO_3 aqueous solution, and rinsing in water	4.2	1.6
After cold rolling through "clean" rolls	4.2	*
After annealing 60 min at 600°C, cleaning in 2% HF–15% HNO_3 aqueous solution, and rinsing in water	*	0.6

*Not determined.

the surface of a fabricated fuel plate that has been cleaned by pickling is contaminated with 0.6 ppm U^{235}, which is nearly a factor of 10 higher than the U^{235} content of normal aluminum stock. This can be accounted for by diffusion of uranium from the fuel core to the surface of the aluminum-clad plate. Calculated values based on preliminary data indicate that diffusion after as little as 4 hr at the 600°C processing temperatures can account for as much as 2 ppm U^{235} at the surface of the aluminum cladding.[34]

The fuel elements of the ORR (which is capable of operation at 30 Mw, a factor of 10 higher than the LITR) have apparently contained less than 3 ppm U^{235} at the fuel plate surface. This conclusion is based on the activity of the radionuclides Na^{24}, I^{133}, I^{131}, and Np^{239}, which are monitored periodically to determine their levels in the active lattice of the reactor. Data representative of the September 1960 sampling are tabulated in Table IV-5 and compared with data from sampling of the LITR. It is apparent that Na^{24} is the major contributor to the activity in both reactors. The reduction in the I^{133} values for the LITR coolant over those previously obtained by Moeller and Leddicotte[32] indicates that the surface contamination resulting from fabrication has been reduced by a significant amount.

Table IV-5 COMPARISON OF RADIOACTIVE NUCLIDES FOUND IN SAMPLES OF LITR AND ORR COOLANT AS OF SEPTEMBER 1960

	Activity, μc/ml	
	In LITR	In ORR
Na^{24}	7.5×10^{-3}	3.2×10^{-2}
Np^{239}	2.1×10^{-4}	6.2×10^{-5}
I^{131}	1.8×10^{-5}	7.0×10^{-5}
I^{133}	1.9×10^{-4}	5.9×10^{-4}

The I^{133} activity in both the LITR and the ORR can be traced to U^{235} surface contamination of approximately 1.25×10^{-7} g/dm^2, which is probably the minimum that can be expected on fuel elements that require heat-treatment at 600°C. Although no maximum limits of the tabulated radionuclides have been established for either ORR or LITR operation, I^{133} activity above 5×10^{-3} μc/ml would be alarming.

The majority of the available information on surface contamination of fuel elements has been obtained for tank type reactors where fission-gas permeation into the atmosphere during reactor operation does not appear to be a major source of difficulty. On the other hand, evidence of activity in the atmosphere above pool reactors does exist. Champion,[35] in analyzing air activity produced by the 1-Mw BSR, found that the level of airborne activity 40 ft from the reactor, as monitored by equipment that measured activity trapped on a filter paper, was a few thousand counts per minute at high power levels and displayed a half life of about 30 min. This airborne activity represented an appreciable portion of the tolerance level. These studies indicated that N^{16} was the largest contributor, and Champion concluded that no fission products were escaping from the fuel elements into the water. The implication was that all the air activity could be accounted for by activation of impurities and dissolved gases in the water. Around a pool reactor operating at 1 Mw or less, the air pollution by gaseous fission products is apparently negligible if the uranium contamination on the fuel plate surfaces is low. On the other hand, for a 5-Mw pool type system, the U^{235} contamination on fuel-element surfaces may cause a hazardous condition.

It is apparent from an examination of the effects of U^{235} contamination on the surfaces of fuel plates that, depending on the application, such contamination can contribute significantly to the problems of operation and maintenance of a reactor because of the health hazards associated with the generated fission products. In all reactors, maintenance is hampered to varying degrees by the radioactivity produced in the system, and, although the contribution from uranium contamination on the surfaces of fuel elements may be small, over a long period of time it may increase the maintenance difficulties. In the case of submersible reactors, the combination of confined space, recirculating air system, and the cited contamination may present serious difficulties. Similarly, the maintenance of pool type research reactors, particularly those operating at 5 Mw, will be more difficult if the surfaces of the fuel elements become unduly contaminated with uranium during fabrication. It is therefore desirable that contamination by uranium on the surfaces of fabricated fuel elements be at a reasonable minimum. The reactor engineer and the fuel-element fabricator can do much to resolve these problems on the basis of a full understanding of their mutual problems.

(R. J. Beaver)

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Section V

CONSEQUENCES OF ACTIVITY RELEASE

Personnel Monitoring

Until recently the methods and equipment available for warning individuals of radiation hazards and for measuring the doses which they may have received were considered to be generally adequate. Present opinion suggests, however, that we should improve methods for (1) internal dosimetry of alpha-particle-emitting radionuclides, (2) low-level neutron dosimetry, (3) gamma-ray dosimetry for doses greater than 500 rads, (4) detection of particulate-associated alpha-particle emitters in the presence of radon daughters, (5) dosimetry for those involved in criticality accidents, and (6) individual dose-rate warning. There is also a growing need for published texts or references devoted primarily to personnel monitoring instrumentation and its relation to health physics.

Internal Dosimetry of Alpha-Particle-Emitting Radionuclides

Whole-body counters, particularly those using gamma-ray spectrometers, can be calibrated for the quantitative determination of internally deposited gamma-ray-emitting radionuclides. Therefore they can be used to measure body burdens of those alpha-particle-emitting radionuclides which decay with associated gamma radiation of sufficient abundance and energy. These include some, but not all, of the alpha-particle emitters that are of greatest concern as internal hazards, i.e., some of the isotopes of uranium, thorium, and plutonium.

A method has been reported by Cofield which permits assay of less than the permissible body burden for uranium.[1] This alpha emitter is potentially responsible for more exposures in the atomic energy programs than any other single element, so a good method for dosimetry is desirable.

The assay of thorium is more difficult. Another study by Cofield[2] shows that the use of the whole-body counter for this determination is restricted because of the complex gamma-ray spectra of thorium and the variability of the ratio of Th^{232} to Th^{228}. Unfortunately, at the present state of the art, the whole-body counter cannot be used to detect some of the most hazardous alpha-particle emitters, such as Pu^{239}, because the associated gamma rays are insufficient in quantity or energy.

Radiochemical analysis of body fluids remains the most widely applied method for the detection and estimation of internally deposited, nongamma-emitting radionuclides. Most of the available useful information about analyses for uranium was reviewed in a symposium[3] held in New York in October 1958. The published report of this symposium is a basic reference for health-physics problems related to work with uranium. Other techniques for determining internal dose, including urinalysis procedures for detecting Pu^{239} and enriched uranium, were described and evaluated at the 1959 Bio-Assay and Analytical Chemistry Meeting.[4]

Small Doses of Neutrons

Neutron dosimetry of dose rates less than 150 mrem/week is complicated by the lack of sensitivity of most emulsions and the instability of nuclear tracks in the latent image. These limitations are true of most film currently available for personal dosimetry of fast neutrons for doses less than 10 rads. However, Cook of Harwell has shown that Ilford C2 emulsion may permit detection of 100 mrem or less and that latent-image fading may not exceed 10 per cent in 12 weeks.[5]

High Doses of Gamma Radiation

Most personal dosimeters for recording gamma radiation use photographic emulsions as detectors. High-range films and special processing techniques to extend the dose range have been in use for several years, but the accuracy of film for dosimetry in these higher ranges is, in practice, not nearly so good as for lower dose levels.

The upper limit of film dosimetry has been determined by the exposure which produces in the developed film a transmission density of 3.0. Therefore a film expected to have, after normal processing, a developed density greater than 3.0 must be processed by a special method that will produce a net density between 0.1 and 3.0 in order that a dose may be determined. This requires foreknowledge of the approximate dose.

A new method for extending the dose range of dosimeter film emulsions to film densities greater than 3.0 by use of X-ray spectrometry has been designed and is in use at Hanford.[6] This method, which does not require changing the normal method of developing the film, should permit better dose estimates. X rays are used to excite the silver atoms on the developed films, and, since the quantity of metallic silver in the developed emulsion is directly proportional to the dose absorbed in the emulsion, the quantity of photoelectric X rays from the silver is proportional to the absorbed dose. By using a spectrometer which counts only the $K\alpha$ X rays, the upper limit of measurement using Du Pont emulsion 510 was extended from the normal 40 rads to as high as 2000 rads.

Small rods of metaphosphate glass are changed by gamma radiation in proportion to the dose received, and this change can be measured with a special reading device.[7] A dosimeter that uses these metaphosphate glass rods has many desirable characteristics, among which are small size (1 mm in diameter, 6 mm long), almost negligible weight, resistance to solvents and moisture, and the stored dose information is not lost in the reading process. The usable dose range is from five to several thousand roentgens, and, although the measured dose is independent of dose rate, a correction is required for energy dependence. These dosimeters are in use at the Union Carbide Nuclear Co. plants at Oak Ridge, Tenn., and Paducah, Ky. They are applicable to personal dosimetry, medical and biological measurements in vivo, process and equipment monitoring, etc. If the stored information can be proved to be stable over long periods, the metaphosphate glass rods will be excellent integrators for "lifetime" doses.

Alpha-Particle Monitors

Monitoring air for those alpha-particle-emitting radionuclides which are more hazardous than Rn^{222} and its daughters is complicated by the universal presence of radon in the open air near the surface of the earth. The concentration of radon and its daughters in the air varies both with geographical location and with the weather, and indoors it depends on the construction materials which enclose it. In some locations it may vary by a factor of 10 within a few hours.

The maximum permissible level (MPL) for Pu^{239} in air produces alpha particles at a rate of 4.4 dis/(min)(m^3); the MPL for Rn^{222} plus its daughters is much larger, being 6.6×10^4 dis/(min)(m^3 of air). At ORNL the alpha-particle-emission rate from radon daughters collected on a filter may vary in a few hours from less than 30 to more than 400 alpha particles/(min)(m^3 of air filtered). Thus, in order to monitor air for Pu^{239}, it is necessary to measure a small quantity of one material in the presence of a large and variable quantity of a similar material.

There are two detectable differences between the radioactivity of Pu^{239} and that of the daughters of Rn^{222}: (1) the alpha particles are of significantly different energies and (2) the decay of the radon daughters is accompanied by beta-gamma activity, whereas the decay of plutonium is accompanied by the emission of only a few weak X rays. Several air monitors have been designed which make use of these differences.

A novel solution to the problem has been developed at Hanford[8] and is based on the facts that (1) the rate of radon deposition on a filter, as measured by the alpha-particle emission from the filter, never exceeds a certain maximum which has been determined for the Hanford region and (2) there is no maximum to the rate of plutonium deposition. The rate meters associated with the alpha detector have been designed to actuate an alarm when a specified rate of increase in counting rate is exceeded. The detector employs zinc sulfide and a photo-

multiplier. This method has the advantage of requiring only a single detector and two rate meters. Under the conditions of radon concentration tested, the instrument can warn of increases in the plutonium concentration of 75 MPL in 18 min or 1000 MPL in less than 2 min.

Dosimetry in Criticality Accidents

Accurate dosimetry was not available when, in the past, individuals were exposed to large doses of radiation in criticality accidents in the United States. There are now systems and methods for such dosimetry, including a rather thorough system developed by the Health Physics Division of ORNL.[9] Neutron dosimetry in this system employs the activation analysis of a series of materials selected for their neutron-energy activation thresholds. These materials are packaged in small compact containers, which include a foil of each type, and the containers are placed in locations wherever the materials requisite to a criticality accident exist. Metaphosphate glass rods and the Union Carbide Nuclear Co. badge meter[10] are used in addition to provide personal dosimetry, and the blood of persons exposed to neutrons is subjected to activation analysis for Na^{24}.

The badge meter contains gold foil for determination of the thermal-neutron flux and a pellet of sulfur for determination of the flux of high-energy neutrons. Exposure to intermediate-energy neutrons may be determined by comparing the data from the badge with data from a nearby dosimeter containing the complete series of activated materials. Direct measurement of an indium foil in the badge permits quick identification of those individuals who have received significant exposure to neutrons. The badges also contain films and metaphosphate glass rods for the determination of the gamma-ray dose received.

Personal Radiation Monitor

The permissible average dose rate of 5 rem/year corresponds to a daily average dose rate of 20 mrem/workday. The common types of ionization-chamber personal dosimeters have a full-scale reading of 200 mr, and therefore the usually acceptable 10 per cent error is equivalent to the permissible average daily dose. A recently developed pocket-size direct-reading dosimeter, which measures 0.9 by 2.6 by 6 in. and which weighs 10 oz, is somewhat more sensitive.[11] This instrument uses an ionization chamber which is essentially energy independent for radiations from 50 kev to 2 Mev. The charge on this chamber controls the flow of current through a 50-μa meter which reads zero when the chamber is initially fully charged and full scale after the chamber has been exposed to a dose of 20 mr.

A new type of instrument developed at ORNL for detection of radiation is called the Personal Radiation Monitor (PRM).[12] This device is about the size of a fountain pen and weighs only $3\frac{1}{2}$ oz. It is worn in the pocket where it emits an audible warning tone and flashes a small neon lamp in the presence of a gamma-radiation field.

The pitch of the warning tone and the rate at which the lamp flashes are directly proportional to the gamma-ray dose rate over a range from background to about 1 r/hr, at which rate a limiting pitch of 2800 cps is reached. By simple modification the instrument can be used to measure dose rates up to about 200 r/hr. The instrument is powered by a 4-volt mercury battery which provides one month of continuous operation. In the interest of safety, there is no on-off switch.

The PRM uses a miniature Geiger-Mueller (G-M) counter as a radiation detector (three transistors and five semiconductor diodes in the circuit) and a tiny hearing-aid earphone coupled to a resonant air column as the source of the warning tone. Temperature compensation is provided over the range from 0 to 120°F. Humidity protection is secured by complete encapsulation of the sensitive electronic assembly in plastic epoxy resin.

Prototype models of the PRM have been used at ORNL for more than a year. Their principal application has been as an immediate warning of unexpected radiation levels, as when an employee oversteps the shadow of shielding or enters a beam of radiation inadvertently, and in the detection of surface contamination and flaws in shielding.

Alpha Contamination Monitoring

An instrument that monitors hands for alpha and beta-gamma contamination simultaneously and at the same time monitors shoes for beta-gamma contamination has been designed and tested at Hanford.[13] The detectors in the hand monitors are 4- by 8-in. sheets of terphenyl

covered with a layer of zinc sulfide and a 1.1 mg/cm^2 light barrier. These detectors are coupled to an RCA 6372 phototube. Two detectors between which the hand is inserted during counting are provided for each hand. Pulse-height differentiation separates the relatively large pulses due to alpha particles from the relatively small pulses due to beta particles. There are alpha and beta registers for each hand, a beta register for the feet, and a speaker for an associated external alpha probe. The circuitry is transistorized, and, according to the author, the instrument should require almost no maintenance.

Prototypes of two portable alpha-contamination monitors have been developed at Chalk River.[14] One has a detector with a sensitive area of 175 cm^2, an average detection efficiency of 16 per cent for Am^{241}, and a background rate of 1 count/min. The other has a detector with a sensitive area of 18 cm^2, a detection efficiency of 13 per cent for Am^{241}, and a background rate in the order of 1 count/hr. Both units are compact and appear, from photos, to be relatively light in weight.

Texts on Health Physics Instruments

Health Physics Instrumentation, a recent text by Handloser,[15] is probably the best basic reference published to date on radiation monitoring for personnel protection. The subjects treated are:

1. Types of radiation and units and limits of radiation exposure.
2. Radiation detectors.
3. Portable survey instruments.
4. Fixed monitors.
5. Neutron monitors.
6. Counting equipment.
7. Calibration techniques.
8. Miscellaneous instrumentation.

Nuclear Radiation Measurement by Sharpe[16] is a valuable concise treatment of much the same material. (E. D. Gupton)

Instruments for Environmental Surveys

Attempts to study or map the distribution of radioactive materials in the general area surrounding a nuclear research or production establishment have brought up special problems in instrumentation not normally present in laboratory work. The instruments must work in the environment they are measuring and are subject to levels of contamination at least as high as the levels of activity being studied. Because the distribution of activity in the environment is usually complex, many samples or readings are typically required in comparison with those required in connection with controlled experiments in the laboratory. The levels of activity are in general very low but are characteristically subject to wide and unpredictable variations. Thus the instruments must be both sensitive and versatile. If they are to be moved about, they must also be rugged and light in weight, and, because they may be exposed to the weather or other dangers, they should, if possible, be inexpensive. In the laboratory, shielding can be provided by thick walls of lead or concrete; for field work this may not be provided.

Some of these problems may be avoided by collecting samples in the field and taking them into the laboratory for analysis. This procedure, however, introduces problems of its own. Many more samples are required, there is inevitable delay, and significant elements of the problem may be missed.

Airborne Radioactivity

Many types of instruments are used for detecting radioactive particles in air, but most are based on the collection of the particles on some type of filter. Discussions of the filters used to remove particulate matter from gases have been presented.[17–19]

One of the chief problems in air monitoring is to obtain, in a reasonable length of time, a sample large enough for analysis. Suppose, for example, that it is desired to detect 10^{-13} μc/cm^3 of beta emission (off-site MPC$_a$ for unknown mixtures of beta emitters) with a detector (end-window G-M counter) that is 10 per cent efficient for betas of the energy present. If the counter has a background of 20 counts/min and if the count rate is to equal the background, the activity collected in the filter must be 200 dis/min or 9×10^{-5} μc. This will require the filtering of 900 m^3 of air, i.e., more than can be handled easily with most field equipment. With the latest expensive, low-background, anticoincidence beta counters, it is possible to achieve backgrounds of 1 count/min

and a geometry approaching 50 per cent efficiency and so reduce the required sample of 9 m^3. But even a blower to sample 9 m^3 through a filter paper of reasonable dimensions in a period of 10 min (32 cfm) will require a motor of approximately ½ hp, and, although this presents no serious problem if 115-volt power is available, the complete instrument obviously will be too heavy and bulky to be used conveniently in the field.[17,20]

For the measurement of extremely low levels of airborne particulate radioactivity in the environment, the samples must be large and the sampling time long if the assay is to be made by standard laboratory type equipment. If sample volume is small, low-background, high-efficiency, laboratory type counters and long counting times must be employed.

Where the requirements are not so severe, as for spot checks to indicate the presence of abnormally large concentrations of radioactive particles, hand-operated, battery-operated, or vehicle manifold-vacuum-operated filter samplers are sometimes used. Where the length of the sampling period is of no consequence, samplers with flow rates from a few liters per minute to a few cubic feet per minute are used, and the filters are removed and counted perhaps weekly.

Ideally, the environmental airborne-radioactivity monitor would sample the air and measure the radioactivity simultaneously, but for very low activity levels, a considerable volume of air must be sampled, and expensive, heavily shielded, laboratory type counters must be employed. If the cost can be justified, large blowers and laboratory type counters (perhaps requiring several detector heads viewing a large filter area) can be used at fixed installations in the field for continuous low-level monitoring, but at present the best method seems to be collection in the field and analysis in the laboratory.

For cases where extreme sensitivity is not required, a number of instruments are in use which both sample and monitor continuously. In one instrument a G-M tube is inserted into a slotted cylinder around which is wrapped a filter paper through which the sample is drawn.[21] This type of instrument indicates the integrated particulate load over the sampling period. Other models use continuously or intermittently moving paper tapes with separate beta-gamma or alpha detectors to provide a more nearly continuous record of particulate activity.[20–22] In these instruments, however, there is commonly a lag between time of collection and time of counting which limits their use as alarm devices.

Nearly all types of air monitors are subject to high background readings produced by the short-lived particulate daughters of naturally occurring radon and thoron which are always present in environmental air in concentrations that vary with meteorological conditions. Several methods have been developed to overcome this problem. One method makes use of the fact that the natural radon and thoron decay products are primarily associated with particles below 1 μ in diameter, whereas man-made particles tend to be larger.[23] In one instrument based on this method, a Vaseline-coated moving tape is used to trap the larger particles in preference to the smaller ones.[24] Another instrument makes use of the fact that the ratio of alpha to beta-gamma activity of the radon and thoron daughters remains constant.[23,25] In this type of monitor the alpha count, the beta-gamma count, and the ratio of the two are continuously monitored. Any appreciable change in the ratio is an indication of the presence of man-made radioactive particles.

All the automatic instruments are complex and costly, approximately $5000 each, and they require regular service and maintenance. On the other hand, it is sometimes even more expensive to obtain and process numerous grab samples, and therefore the automatic instruments will no doubt see increasing use.

Some sampling instruments used in environmental monitoring separate the sample into fractions on the basis of particle size. One of the most widely known of these instruments, the cascade impactor, has a low flow rate[20,26] of only 17½ liters/min. A larger two-stage instrument with a flow rate of up to 40 cfm contains a cyclone separator and a high-volume filter sampler.[27,28] This instrument can be adjusted to give a first-stage fraction that is representative of particles which would be retained in the upper portions of the human respiratory tract and a filter fraction which represents the particles that would penetrate to the lower reaches of the lungs. Such an instrument provides a more realistic evaluation of the true hazard associated with airborne particles.

Fallout trays, flat sheets, or plates coated with some type of adhesive are widely used in environmental monitoring.[29] Although they are not accurate indicators of the airborne activity, they do provide a general index and, when used in conjunction with filter samplers, provide valuable supplementary information. Modified rain gauges are also generally employed in environmental monitoring to collect rainout samples and provide some information about airborne particulate activity.[21,29]

When specific radioactive gases are a problem, specialized gas samplers are used in environmental monitoring. For example, for iodine monitoring, alkaline scrubbers or small activated-carbon cartridges are used to collect samples. One instrument consists of a small activated-carbon cartridge inserted into the well of a scintillation counter.[30] Air is drawn continuously through the cartridge, and thus the counter indicates the integrated iodine concentration. At one installation, where Ar^{41} is the major radioactive gas produced, an instrument is used which consists of an annular beta window proportional counter surrounding a 7-liter sample chamber.[31] The whole chamber is buried to prevent reading direct radiation, and the continuously flowing sample stream is prefiltered to remove particulate activity.

On those particle- and gas-sampling instruments in which the radioactivity counters are not an integral part of the instrument, samples are counted in the laboratory on standard instruments, such as end-window G-M counters, internal proportional counters, alpha scintillation counters, and gamma scintillation counters and spectrometers. Where the number of routine samples is large, automatic sample changers are used.[32] In addition to counting of samples, autoradiographic techniques are frequently used to determine numbers and approximate activity of radioactive particles. Typically, standard X-ray film and development procedures are used, and counting and sizing of spots are done by eye on a light table.[29]

Radiation Level

Film badges and ionization-chamber dosimeters of the types used for personnel monitoring can be left out in the field to provide routine integrated-dose measurements.[21,22] The film badges have a system of filters built into the film holder which makes it possible to distinguish between deep penetrating and moderately penetrating beta and gamma radiation, as well as the dose produced by neutrons of various energies.

For instantaneous measurements, many types of G-M tubes, proportional chambers, ionization chambers, and scintillation crystal instruments, both portable and fixed, are used. Fixed stations for radiation-level monitoring frequently employ an exposed G-M tube. The scintillation detector is more sensitive than the G-M detector and is better suited for low-level measurements, but it requires greater protection and is somewhat more expensive. The scintillation counter, because of its greater sensitivity, is also becoming increasingly popular for portable alpha and beta-gamma survey instruments. Because a very thin window must be used for alpha measurements, the problem of maintaining a lighttight instrument in the field can be a disadvantage.[33] However, other types of alpha counters have their own peculiar problems, and, in comparison, the scintillation counter is still attractive.

The development of the transistor and other semiconductors has made possible the development of lighter, more compact, and more reliable portable instruments. The much lower power requirements of transistors, as compared with vacuum tubes, have made it possible to provide much more sophisticated circuitry in portable instruments.[33,34] However, transistorized instruments use the same or similar detectors as vacuum tubes, and, since the detector determines the response of the instrument to the various types of radiation, the improvement is a matter of degree rather than principle. The semiconductor detector appears to hold much promise as a monitor for alpha and neutron radiations because the power supplies will be simple, the detector is rugged, and size and price will be smaller than for a comparable scintillation detector. However, the usefulness of the semiconductor detector for beta and gamma monitoring is not yet certain.[35]

One of the problems faced in the use of the standard radiation-level monitors that normally measure only background levels is their limited range which may not be sufficient to cover unusual conditions. For example, the standard portable G-M survey instrument (a pulse type detector) will usually have ranges of 0.2, 2, and 20 mr/hr (radium calibration). Above these levels the instrument begins to saturate and

may even begin reading downscale as the radiation is further increased, thus giving a false indication of decreasing radiation level. Several new instruments have been developed in which a single detector covers a wide range of radiation levels. For example, a new halogen-quenched G-M tube operates as a pulse type detector from background up to 0.1 r/hr and as an ionization chamber[36] up to 50 r/hr. Readout is on a logarithmic scale. Another instrument covers the range from 1 μr/hr to 1000 r/hr on 10 automatically switched linear ranges.[37]

Where large areas are to be monitored by fixed stations, automatic telemetering systems are being used increasingly. Some of these use a single pair of wires to send power to the remote instrument and to return the signal to the central station.[37,38] With a system of this type, it is possible to locate fixed stations in areas where 115-volt power is not available and to have the information from the entire area automatically presented at the central station.

It is now common to monitor large areas with portable instruments, usually scintillation counters, carried in land vehicles, helicopters, and airplanes. One vehicle-mounted instrument developed in England is a scintillation counter with ranges of from 15 μr/hr to 1.5 mr/hr, which is operated from flashlight batteries.[39] The entire instrument is attached to a 10-ft pole mounted vertically on the front bumper. Such an installation gives a greater area view, less sensitivity to road surface anomalies, and less effect from vehicle contamination than if the instrument were mounted in or directly on the vehicle. The recommended vehicle speed is 20 mph. An even greater area may be surveyed with an instrument mounted in an aircraft. Such monitoring is less dependent on localized anomalies, and it is reported that aerial surveying is a valuable technique when the ground contamination produces a dose of about 75 μr/hr at a height of 1 m above the ground.[40] This corresponds to about 8 μc of Cs^{137} per square meter. Another report indicates that 1 μc of I^{131} per square meter may be detected with portable scintillation equipment in light aircraft.[25]

Terrestrial and Aquatic Biota

Biota samples, such as small mammals, fish, arthropods, leaves, and grass, are generally taken by standard methods such as trapping, netting, seining, and cutting. Additionally, thyroid, bone, and other samples from domestic animals may be obtained from slaughter houses.

In situ measurement of fauna is generally not practical. In some cases, e.g., gross surface contamination, it might be possible to perform *in situ* measurements on flora; usually, however, the radioactivity level of the samples is quite low and would require the use of low-level laboratory type counters with their attendant heavy shields which are scarcely portable. In fact, even for counting on the best laboratory counters, it is frequently necessary to concentrate the activity by techniques such as ashing or digesting, particularly if the nuclides being monitored are alpha or beta emitters.[41,42]

Gamma-emitting nuclides offer possibilities for monitoring of bulk samples. For example, it has been suggested that a G-M counter inserted into a churn of milk is sufficient to detect contamination at levels considerably lower[43] than the MPC. Such a technique could be useful following an incident such as the Windscale accident.

For gamma counting and spectroscopy of bulk biota samples, the thallium-activated sodium iodide crystal is in widespread use. A 3- by 3-in. or 4- by 4-in. crystal, either well or end-window type, is commonly used.[30] For extreme sensitivity, crystals such as the $9\frac{3}{8}$-in.-diameter $8\frac{1}{2}$-in.-thick crystal in use at Hanford are desirable. This crystal has a $3\frac{1}{4}$-in.-diameter well 5 in. deep. It is reported[44] that a 50-min count of a 500-g sample gives a detection limit (99 per cent confidence level) on the order of 2×10^{-8} μc/g for I^{131} or Cs^{137}.

For materials such as grass, sample containers that slip over a crystal assembly and thus surround the crystal with samples have been fabricated. Counters designed for large-volume environmental samples require large shields to accommodate such samples. One version, referred to as a "pickle barrel," has internal dimensions of 12 by 18 in., and its lid is operated by a motor-driven hydraulic jack.[45]

Although the large-crystal scintillation detectors and associated large-volume shields are rather expensive, they appear to be the most promising avenue for identification and measurement of gamma-emitting nuclides in bulk biota samples. The appearance on the

market in the past several years of a number of completely transistorized multichannel gamma-spectrometer analyzers has further enhanced the application of gamma spectrometry to environmental samples. Although they are not cheap (~ $10,000), they are perhaps an order of magnitude more compact and more reliable than vacuum-tube types. (At least one model is designed for battery operation in the field, but, for low-level samples, the detector shield requirements drastically limit its otherwise excellent portability.)

Ground and Surface Water

Some indications of ground-water hard beta and gamma radioactivity can be obtained by specially designed well loggers with either G-M tube or sodium iodide crystal detectors.[46,47] Once appreciable activity has been detected in a well, the logger may no longer be a reliable indicator of water activity, since the soil itself will become contaminated and may dominate the instrument reading. As in the case of other types of environmental monitoring, if it is desired to detect soft beta and alpha activities or MPC_w levels of the more hazardous nuclides, it is necessary to resort to laboratory type counters and/or various means of sample concentration. Standard instruments for soft beta and alpha emitters depend on placing the sample inside the detecting chamber or viewing the sample through a very thin window which cannot withstand the pressures encountered with immersion type well loggers. With the new semiconductor detector, it may be possible to overcome this obstacle.

Ground water is frequently monitored by grab samples taken from drilled wells. A well may be pumped or specific-depth samples taken by means of stoppered bottles controlled from the surface. However, neither of these methods produces a sample that can be accurately related to a specific depth in the soil. One method which has been developed in an attempt to overcome this difficulty consists of the use of Alundum thimbles connected to the surface by polyethylene tubing. After placement of the thimbles at the proper depth, the well is then backfilled. Another development is the porous bronze piezometer which can be driven into the soil in areas where rock is not encountered.[46]

It should be noted that, in preparing grab samples for laboratory counting, there is a limit to the amount of sample concentration which is advantageous. Sample concentration is limited by self-absorption of the radiation in the final sample itself. In the case of alpha and beta measurements, the dissolved solids present in even the purest ground water may limit the optimum initial sample to a few liters. In addition, in the process of evaporation, some of the activity may be lost through volatilization and plating-out on the surface of the evaporating vessel.

Stream and river monitoring is done by grab sampling, flow-proportional sampling, or *in situ* continuous radioactivity monitoring. Existing continuous beta-gamma monitors of the instantaneous type will indicate concentrations[48] of Sr^{90} down to about 10^{-5} μc/ml, or 100 times MPC. Another approach is the integrating continuous monitor which uses an ion-exchange resin to concentrate the activity. In one version it was found that 10^{-3} μc of beta activity on the resin could be detected readily. Thus the MPC level of Sr^{90} could be detected after passing 10 liters of water through the resin.[23]

Because of their short ranges, continuous monitors for alpha and soft-beta radiations do not appear to be feasible in low-level applications using standard detectors, and thus the sampling methods must be employed.[23] River- and stream-sampling instruments are generally of standard types, such as the Trebler flow-proportional sampler and various versions of spot and integrated-depth grab samplers.[21,25,28] None of the existing methods of sampling are entirely adequate, the chief problem being that of obtaining a sample which is both representative of and proportional to the total flow. If the entire stream can be channeled through a weir, then the Trebler type sampler, which uses a scoop shaped to match the characteristics of the weir, will obtain a sample that is proportional to flow and is reasonably representative of the flow at the time of sample collection; however, since sampling is not continuous, slugs of activity in the water may be missed.

The same considerations concerning low-level laboratory instrumentation apply for environmental water samples as for other environmental samples. Samples may be concentrated, or, in some cases, large volumes may be counted directly in the large-sample scintillation counters.

Soils and Sediments

Soil and river- or stream-sediment samples are counted in the laboratory on the same instruments as those used for other environmental samples. Bottom sediments are collected with dredges, driven tubes, and other standard devices which are not entirely satisfactory from the standpoint of obtaining undisturbed samples. Suspended sediments are generally filtered from water samples.[21,29,49]

One instrument has proved quite valuable for *in situ* spot checks of hard-beta and gamma radioactivity in bottom sediment and is called the "Flounder." It consists of a bank of 12 G-M tubes in a watertight probe connected to a portable scaler.[49] The use of an instrument of this type can appreciably lower the number of samples that must be processed in the laboratory. Although it will give no indication of the depth distribution of activity in the bottom sediment, extensive soundings with this device coupled with a few well-chosen grab samples can give a reasonably accurate picture of bottom-sediment activity which could only be obtained at great effort and expense with laboratory counting of grab samples. Since the water itself is an effective shield, it might be possible to obtain additional information *in situ* through the use of a large sodium iodide crystal and portable gamma spectrometer. However, the problem of leakproof packaging of the crystal, phototube, and possible circuitry required in the probe would probably be greater than in the case of the G-M tube. Due to the much greater value of the probe components, the reliability of the watertight seal would be more critical.

Conclusions

In the field of environmental monitoring, there is yet much to be desired in the way of instrumentation. Some recent developments, such as the large sodium iodide crystal gamma detector, the low-level beta counter, and the transistorized multichannel gamma-spectrometer analyzer, have done much to improve the speed and accuracy of laboratory analyses of environmental samples. Although some improvements in *in situ* monitoring, such as transistorization of standard instruments, applications of scintillation detectors, remote station telemetering, and vehicle and aerial surveying, have been made, the greatest current need for further development appears to lie here. Perhaps the semiconductor detector will meet this need.

(R. L. Bradshaw)

Public Relations Following Environmental Contamination

Associated with every nuclear incident is the problem of informing the public of what has happened. This problem has always existed, but, during the early stages of the nuclear age, the release of information was restricted by the classified nature of atomic energy work and could therefore be controlled. With the widespread use of atomic energy in its various forms in industry during the last decade and the accidents which have inevitably occurred, the problem of public relations has assumed renewed importance.

Although any of several incidents could be used to illustrate the problem, only four incidents will be referred to here. These four are well known because of the publicity they have received, and they illustrate well the development of public relations in such incidents and the Commission's concern for their proper handling. The four incidents to be referred to are the Sylvania pyrophoricity incident,[50] the Kellogg incident at Houston,[51] the Y-12 criticality incident,[52] and the SL-1 reactor incident.[53] (Since this article is principally concerned with public relations relative to these incidents, rather than the incidents themselves, the reader is advised to refer to the references cited for additional information on the incidents.) The first of these incidents serves to highlight the problem, and the remaining three illustrate the development in public relations since that time.

Hayes of the AEC Health and Safety Branch has referred[54] to public relations as one of the most difficult problems raised by accidents in nuclear energy installations. Certainly the inadequate program for the dissemination of information following the Sylvania pyrophoricity incident, as described by Eisenbud,[50] contributed significantly to the misinformation, confusion, and general hysteria that developed. Some of the lessons learned from this incident were discussed by Eisenbud and were subsequently enlarged upon by Salisbury.[55] Principal among these are the virtue of "careful, explicit and frequent reporting." This incident also brought to light the importance of radio and television

in informing the neighborhood residents and the need to use well-understood unambiguous terminology.

AEC has no control, in regard to public relations, over outside organizations engaged in atomic energy endeavors, but an accident in Houston[51] early in 1957 brought the Commission into the picture because of the resultant poor public relations. During a decladding operation at the M. W. Kellogg plant, irradiated pellets of Ir^{192} and aluminum metal powder were broken. These contaminated the laboratory, and several employees went home with their clothing contaminated. There was considerable concern in the community because of this spread of activity. Subsequent reports[56] revealed that the exposure was only slightly in excess of permissible limits and that there was no evidence of significant ingestion of radioactive materials. One employee claimed, however, to be ill as a result of the accident, and his case received nationwide attention for an extended period. The situation became a national *cause célèbre* to the extent that a special appropriation was asked in Congress for the benefit of the victim.

In an effort to help clarify the situation, AEC offered to pay for a full evaluation of the medical status of Jackson E. McVey and his family.[57] The Commission's letter said:

> Since you are unwilling to be examined at the Baylor University College of Medicine (Houston), we are prepared, as indicated in our letter of September 12, 1960, to arrange for a complete diagnostic examination at another clinic, namely the Mayo Clinic in Rochester, Minnesota. You, your wife and your two children would be examined at the expense of the AEC. The AEC will also arrange to assume the cost of the travel of your family from Houston, Texas, to Rochester, Minnesota, and return, together with your family's subsistence expenses at Rochester, pertinent to the Clinic's checkup. We will work out the travel and subsistence details with you, if you consent to go.
>
> You and the AEC would be furnished with full reports of the Clinic's findings; the AEC would have the right to use the information and to make it a matter of public record.

The employee subsequently accepted the AEC offer, and the results of the evaluation at the Mayo Clinic were made public in March.[58] The report by Dr. Kiely of the Mayo Clinic said, "I told Mr. McVey that, in my judgment, his physical health at the present time was good and at the time of our examination we could see no specific abnormal physical findings as a result of his previous exposure to radiation."

The poor publicity that attended both of these incidents led to the realization that the hysteria and most of the undue concern which had been evidenced could have been allayed by prompt, factual reporting. Accordingly, when the mid-1958 incident at the Y-12 Plant in Oak Ridge occurred, the community, through several media, was kept advised of information as it became available.[59]

In the incident at the Y-12 Plant, an accidental criticality in a material fabrication plant exposed eight employees to significant levels of radiation. The first emission of radiation activated the plant's alarm system, and personnel evacuated the buildings. Since the plant is less than two miles from the City of Oak Ridge, the accident quickly became public knowledge. The first official statement on the incident was made less than an hour after the accident occurred. For 2 hr following the initial statement, AEC's Public Information Staff handled countless inquiries, sifted fact from rumor in preparing a second statement, and watched community reaction for signs of possible fright or panic. During that period, the community attitude was one of anxiety for the safety of relatives or friends working at the plant. Radio stations reported, however, a few inquiries from residents who asked if they should evacuate.

A second press release, issued 3 hr after the accident, emphasized that there was no danger to the community or surrounding area, and this put an end to rumors. Subsequent statements, in the days and weeks that followed, gave the public details concerning the accident and the conditions of the men who were exposed. The fact that the news was accepted locally with general calmness reflects the importance of dealing fairly and quickly with the public in such cases.

The handling of the Y-12 accident has served as a pattern for dealing with similar incidents at other Commission installations. This format of reporting such news quickly and accurately was utilized during the recent accident at the National Reactor Testing Station at Arco, Idaho.[53,60] Three men were killed on Jan. 3, 1961, when an excursion occurred at the Stationary Low-Power Plant No. 1. Once more, quick relay of detailed information to the public prevented excessive worry about contamination in the immediate areas. In addition, the con-

stant feeding of information to public media has resulted in a fair, unemotional report of the accident in most publications.

The Commission's public information functions are not limited to "putting out fires" in emergency situations. Since the AEC took over the nation's atomic energy program in 1947, its public information staff has been active in informing people, in this country and abroad, of the peaceful uses of atomic energy and its potentialities. A memorandum from the Commission to the General Manager[61] has reiterated AEC's desire "to encourage widespread participation in the development and utilization of atomic energy for peaceful purposes to the maximum extent consistent with the common defense and security and with the health and safety of the public."

Public attention is obviously centered on the possibility of environmental contamination during atomic incidents, since their interest is hardly noticeable when routine data reports show that all is going well at nuclear installations. The AEC, early in 1960, announced arrangements for public reporting, on a regular basis, of radioactivity data[62] collected in the vicinity of major Commission installations. Those reports are produced from routine monitoring programs around plants and laboratories where operations are of such a nature that perimeter radioactivity monitoring surveys are required. Such monitoring is done to check controls and determine effects, if any, on surrounding areas.

Although several quarterly reports of routine monitoring information have been issued, the interest displayed by the communications media has been practically nil. There have been only a few stories in the newspapers serving the 14 areas around nuclear installations. But such disinterest is not unexpected. In the first place, the quarterly reports in most cases indicated that nuclear plants contributed little to air, ground, or water contamination of the surrounding neighborhoods. Unlike nuclear accidents, such "good" news hardly attracts attention, and editors usually bypass this material for something more appealing (or appalling) to the public. Second, the reports are technical enough to deter newspapermen who are unfamiliar with nuclear terms. This, however, should not keep them from at least printing the summary, which is presented in layman's language.

Another AEC memorandum,[63] issued in October 1960, urged AEC and contractor public information offices to give prompt and full reports of accidents and incidents involving:

> ... hazard to the public from radiation or radioactive materials; exposure of workers to radiation or radioactive materials above the permissible amounts for occupational exposure; contamination of more than a minor nature to personnel or property on-site or off-site; and circumstances which would be likely to create rumors or exaggerated reports, in the absence of official issuance of information.

It is evident, therefore, that the Commission is eager to keep the public fully informed about such activities.

Public relations problems of this type are not confined to the nuclear energy field alone. Nearly all industries are besieged with similar emergencies which bring them ill will or disfavor. However, the newness of atomic energy, as well as its association with atomic bombs and the destruction they can create, results in public suspicion and apprehension. To a large extent, these are a by-product of misinformation, or lack of knowledge, about nuclear energy and what it can mean to mankind. A recent survey of public attitudes toward atomic energy, conducted by the General Electric Company, has revealed that, "... public understanding of the facts of atomic energy is directly related to public acceptance of industrial and other nonmilitary applications of atomic energy, while ignorance and confusion go hand-in-hand with fear and doubts of the peacetime role of the atom."

Although public relations activities are undertaken by AEC and its contractors, there is a need for most of the other organizations engaged in nuclear energy work to be aware of the public interest when similar incidents occur. (S. Sinichak)

Education and Training in Health Physics

The development of atomic energy has been accompanied by a phenomenal increase in the problems of radiation protection. Prior to 1942, virtually all radiation hazards were associated with the use of X rays and radium, and expo-

sures were limited to a few radium dial painters following a fixed procedure and to a few doctors, scientists, and technicians using small quantities of radium or working around X-ray machines and small accelerators. Relatively few people had even a remote possibility of exposure to ionizing radiation. During this period, many injuries and deaths directly attributable to radiation exposure stimulated the establishment of codes and standards for radiation protection. In 1922 the American Roentgen Ray Society published its first radiation protection measures. From time to time in the years that followed, the U. S. Advisory Committee on X-ray and Radium Protection recommended maximum permissible doses for external exposure and maximum permissible amounts for radium deposited in the body. These standards, although limited, served as invaluable guides when plans were being made early in 1942 for the construction of the large nuclear reactors at Oak Ridge and Hanford.

With the advent of these reactors, the problem became that of large numbers of persons working in the vicinity of sources releasing radiation, the magnitude of which seemed fantastic and which, in part, consisted of neutrons, whose behavior was little known. Furthermore, in the years following 1946, there have been ever-increasing varieties and quantities of isotopes available. Even a cursory glance at the report of the AEC Office of Isotopes Development shows not only greater quantities and greater varieties of the isotopes being shipped but also the number of users increasing rapidly over the years. Previously it was a matter of a few scientists and medical men using rare cyclotron-produced radioisotopes or the usual radium sources; now there are many scientists in many laboratories working with larger quantities of new materials and applying them to problems in the fields of medicine, agriculture, industry, physics, chemistry, etc.

Simultaneously with the increased use of radioactive materials, there has been an increase in the number of nuclear reactor installations, as well as accelerators, with emphasis on machines capable of accelerating particles to higher and higher energies. Likewise, X-ray machines have been increasing in number and versatility.

All this adds up to a manyfold increase in the problems of radiation protection. The hazards have increased in magnitude over the past 10 years, and they are no longer confined to a small number of persons in a few laboratories and hospitals. The unprecedented problems in radiation hazards encountered in the Manhattan Project were solved by the cooperation of physicists, chemists, biologists, and medical men; a new specialist appeared who is known by various names, the most common being "health physicist"—a specialist whose problem it is to detect, to control, and to protect against radiation hazards.

In the older plants, small efficient groups of trained people developed around an exceedingly small number of individuals who had gained experience in the field of radiation protection prior to the war. In the newer AEC plants, some of the health physicists have been recruited from the staffs of the older plants, whereas others have been trained for the job either through the AEC fellowship program[64] or on-the-job training in one of the established health-physics divisions.

The constantly expanding AEC programs, with new reactor sites and new laboratories; the increased use of radioactive isotopes in industry and in research; and the new problems in civil and military defense, in industrial hygiene, and in environmental health have increased the demand for trained health physicists by the AEC, the U. S. Public Health Service, and the national military establishment, as well as industry, universities, hospitals, and laboratories. Today, in any hospital, research laboratory, or industrial organization using radioactive materials, or operating reactors and/or accelerators, it is essential to have available an individual trained in radiation monitoring and in the handling, measurement, and disposal of isotopes and their associated by-products. No longer can it be assumed that the person handling the radioactive material or the one responsible for the accelerator, etc., knows how to give, or will give, the proper attention to the protection of himself, his coworkers, and his environment. The degree of training needed by the person who is to be responsible for radiation safety will depend on the nature and magnitude of the operations involved.

There is a need for trained people which ranges from the equivalent of the plant monitor to the responsible and highly trained health physicist. In 1949 the AEC undertook support of a fellowship program designed to train college graduates for positions in health physics.

Fellows take courses at the first-year graduate level at a specified university. After the one academic year of university study, a three-month summer field-training period at a government laboratory follows. Upon successful completion of this program, certain Fellows are granted extensions to the fellowship to enable them to earn the master's degree. The thesis project may be done at the university or at a national laboratory.

The universities and laboratories where training under the fellowships may be obtained are as follows:

University of California and Lawrence Radiation Laboratory
Harvard University and Brookhaven National Laboratory
University of Kansas and National Reactor Testing Station
University of Michigan and Argonne National Laboratory
University of Puerto Rico and Puerto Rico Nuclear Center
University of Rochester and Brookhaven National Laboratory
Vanderbilt University and Oak Ridge National Laboratory
University of Washington and Hanford Atomic Products Operation

The Vanderbilt-ORNL program began in 1951. Since its inception, 227 Fellows have completed the program. The academic year at Vanderbilt University is spent on physics, mathematics, biology, biological effects of ionizing radiation, radiological units, instrumentation, external exposure, internal exposure, radioactive waste disposal, radiation protection, and toxicology. The ORNL summer training consists of a 12-week program intended to acquaint the Fellows with all phases of applied work. These include radiation detection, radiation measurement, air sampling, contamination control, use of protective clothing, personnel and area monitoring, and record-keeping. Some time is also spent with the research groups in waste disposal and dosimetry. Graduates of this program are employed in government laboratories, universities, medical installations, public health units, and industries where they are concerned both with research and applied problems in health physics.

It is fairly obvious that the AEC fellowship program is designed to train "senior" men in health physics. In an attempt to provide training for other groups interested in radiation protection, special short courses were instituted which vary in length from one week to 10 weeks. Courses of this type have been given for Army, Air Force, Navy, and Public Health Service personnel, AEC contractors, Oak Ridge School of Reactor Technology (ORSORT) students, and health physics surveyors at ORNL. Three international courses of this type have also been given.

The need also exists for lectures on radiation protection to familiarize all personnel with the problems and to present basic fundamentals that will enable them to do their jobs with a keener awareness of radiation hazards. A program was initiated recently at ORNL by the Radiation Safety and Control Group to fill this need.[65] In previous years this has been done by request for various divisions and also for many of the trades people. A short orientation in radiation protection is included as part of the over-all ORNL orientation program. Lectures have been given on a scheduled basis at the Oak Ridge Institute of Nuclear Studies as part of the medical, isotope, and agricultural programs. Manuals have been used by various groups for short courses in health physics.[66–73] Any one of these manuals could serve as an outline for a short course if it is recognized that problems peculiar to the particular laboratory are stressed.

Similar training programs are available at the large atomic energy establishments abroad.[73] A wide range of courses dealing with health physics and radiobiology is offered at Harwell and Saclay. In radiation protection, as in any service, it is the experienced man who has the trust and respect of the people he serves.

(E. E. Anderson and M. F. Fair)

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Section VI

CURRENT EVENTS

Vallecitos Boiling-Water Reactor Operating Experience Through March 1960

This article is one of a series of *Nuclear Safety* reviews describing the operating experience of particular reactors from the viewpoint of reactor safety. No comprehensive survey of the operating experience of the Vallecitos Boiling-Water Reactor (VBWR) has been published so far. The first two General Electric Company operating reports[1,2] and references 3 and 4 appear to be the main sources of the available information. The operations reviewed here extend from startup in August 1957 to March 1960. Subsequent experience may be recounted when additional operating data become available. Modifications to the plant and the operating procedures made after March 1960 are in most instances not discussed in this article.

The reactor was designed as a developmental and experimental device for acquiring information concerning the design of the Dresden Nuclear Power Station, in particular, and to contribute toward the development of boiling-water reactors, in general. Included among the characteristics to be studied were: "Dual-cycle operation, limitations on the power capabilities of boiling-water reactors, fuel design, fuel burnup, the effects and effectiveness of various types of water treatment, structural materials and mechanical features, and the suitability of boiling-water reactors for production of heat and electrical energy, and for propulsion."[5] The plant consists of a boiling-water reactor that generates steam to drive a turbine generator. Rated operation of the turbine requires a reactor output of 20,000 kw(t). The rated generator output is 5200 kw. The reactor can be operated at greater power by exhausting steam into the turbine condenser or by dissipating heat in the steam generator. The reactor and the recirculating loop are located in a gastight steel enclosure. The turbine-generator plant is housed in an adjacent sheet-metal building. A pressure vessel contains the light-water-moderated boiling-water reactor core, which operates nominally at 1000 psia. A pressure-reducing station decreases the steam pressure to 450 psia at 456°F, the nominal turbine design condition. Normally the reactor is operated in a single-cycle mode with natural circulation. Experimental irradiations are performed in the "open" core of the reactor.

In May 1959, after more than a year of operation, an in-pile loop for demonstrating the basic feasibility of using superheated steam as a reactor coolant was incorporated in the facility. The designation of this loop is the Superheat Advance Demonstration Experiment (SADE). The test loop consists of a single-channel test fuel element isolated from the water in the VBWR core by a process tube located in a regular fuel-element position. Steam for cooling the test element is taken from the control-rod-drive balance-cylinder steam line. The steam passes through a special flange on a refueling port, through the test section, and exits from the reactor through the same refueling port. From there it passes through a flow control valve, where the pressure is reduced, to a spray desuperheater, and finally to a separate condenser. Startup and emergency cooling were available from a nonlubricated air compressor that operates continuously to provide other air services within the VBWR containment building. Exit steam conditions for a typical test element were 775°F at 1000 psia. The superheat power is limited to a maximum of 5 kw(t) at this time.[5] The maximum superheat power during operation[6] is about 60 kw(t).

Operating History

The reactor first went critical on Aug. 3, 1957. Power operation began in the middle of October, and on the 19th of that month the reactor was first connected to the Pacific Gas and Electric Company system. On Nov. 13, 1957, irradiation of the first test fuel element began. By Feb. 7, 1958, the reactor had operated for a total time of about 1100 hr and produced a total electric power[3] of about 1.7×10^6 kw-hr. The reactor was shut down from Feb. 7 to May 3, 1958, for replacement of control rods, as will be discussed later. A main coolant recirculating pump was also added at this time. Operation was resumed May 3, 1958; and by Jan. 30, 1959, the reactor had produced a total of about 1.5×10^7 kw-hr of electricity and 8.5×10^7 kw-hr of thermal power, covering 5600 hr of operation. On Jan. 30, 1959, AEC approved raising the reactor power level from 20 Mw(t) [30-Mw(t) peak] to 50 Mw(t). From Jan. 30, 1959, to Sept. 23, 1959, the total electrical output was about 10^7 kw-hr, and the total thermal output was 6.5×10^7 kw-hr. The reactor was in operation about 65 per cent of the time, and most of the shutdowns were required by the experiments then under way. The Pacific Gas and Electric Company, operator of the electrical plant, estimated the plant availability to be 95.5 per cent during a typical three-month period if shutdowns occasioned by the experimental program were excluded.

On Sept. 23, 1959, the reactor was shut down when it was discovered that the control-rod scram time was longer than set forth in the license application. This ended reactor operation for the period covered by this review, except for a brief testing period at low power. By the middle of December 1959, a previously planned reactor modification program began. This included replacement of the core support structure and the forced-circulation water pump to allow study of high-power density and high-specific-power boiling-water reactor cores.

Radiation Level

In a direct-cycle boiling-water reactor, the primary steam passes directly from the reactor into the plant steam system. Corrosion products, coolant activated in the core, and fission products released from the fuel elements that are entrained in the primary steam are carried to the turbine, condenser, and the remainder of the steam, condensate, and feedwater systems. Operating experience with the VBWR has shown that radioactive contamination of the plant has been considerably less than expected originally.[7] At a reactor power of 30 Mw, the radiation levels in the turbine building have not exceeded 6 mr/hr, except in the vicinity of the high-pressure end of the turbine and the condenser where the levels rose to 15 mr/hr and at the air ejector where contact levels as high as 100 mr/hr have been measured. Gamma spectrometer analysis has indicated that this radiation is principally due to N^{16} activity, with a half life of about 7 sec. Small amounts of NaOH have been added during tests to determine the separation factor on boiling. The separation factor employed here signifies the ratio of NaOH concentration in water to the NaOH concentration in steam. The activity of Na^{24} measured in the reactor water and in the steam indicated a separation factor of 40,000 for Na^{24}. Traces of fission products in the water due to surface contamination of fuel have indicated comparable factors for nongaseous activity, including halogens.[7] The F^{18} resulting from the proton-neutron reaction on O^{18} was found to dominate the cooling-water activity from a few minutes to a few hours after shutdown. The half life of this nuclide is 112 min. High water purity has been maintained by cleaning up the condensate stream with a bypass resin-bed ion exchanger before returning the water to the reactor. The resistivity of the reactor water has generally been maintained at over 1 megohm, whereas the water entering the reactor had a resistivity[7] of over 5 megohms.

After reactor shutdown, radiation levels outside and inside the turbine casing quickly decreased to a fraction of 1 mr/hr. Normal maintenance procedures would therefore be employed.[7]

The reactor modification program begun in December 1959 included changes in the main coolant piping and reactor vessel internals.[4] Thus it afforded a unique opportunity to determine the radiation level of normally inaccessible parts of the plant. After the modification had begun, it was determined that the main 18-in. stainless-steel coolant piping could be cut without extensive system decontamination. Typical radiation exposures for the people doing this work were 30 to 40 mr/day. Removal of the control-rod blades did not require shielding, as had been anticipated. The maximum dose

rate 35 ft from a control-rod blade suspended in air was 1 r/hr. Crane-operator exposure during the removal of the seven control-rod blades was 23 mr. Materials that had operated in the upper steam phase of the reactor vessel had radiation levels of 25 mr/hr or less. The steam baffle in this region had a maximum radiation level of 15 mr/hr at a distance of 4 in. Structure below the water level of the reactor vessel had greater activity levels. The lower grid, located within 5 in. of the fuel, had a maximum radiation level of 7000 r/hr measured within the grid near the vessel center line. Radiation levels within the grid at core edge and 15 in. out from the edge of the core were 2000 and 60 r/hr, respectively. The maximum surface radiation level along the vessel wall was about 20 r/hr at the elevation of the core center line. The level decreased to about 0.5 r/hr at the elevation of the top of the core, increasing again to about 0.7 r/hr at the operating water level. This increase was attributed to deposition of surface contamination. Most of the components had a thin, reddish-brown deposit on them that was probably magnetite. Radiochemical analysis of this deposit showed that the major part of the activity was from Co^{58} and Co^{60}, with lesser amounts of Zn^{65}, Fe^{59}, Mn^{54}, and Zr^{95}. All these isotopes, except Zr^{95}, are thought to have originated in the materials of the reactor system.[4]

Operation with Failed Fuel Elements

A series of failed UO_2 fuel-element tests carried out under a wide variety of conditions has been described.[8] Cladding failures incurred in service as well as intentional failures were studied. After three days of operation at 10 Mw with a grossly defective UO_2 fuel element, there was no residual activity on removal of the element. Despite extended operation with defective UO_2 fuel, there has been no measurable increase in background activity in the reactor or steam turbine system.[8]

Altogether, about 200 g of UO_2 has been lost to the reactor during $1\frac{1}{2}$ years of operation.[4] After the reactor vessel had been opened during the modification program beginning in December 1959, Zr^{95} was the only fission product found to be deposited inside the vessel. Its activity level was low compared with the Co^{60} activity. No evidence of large amounts of UO_2 was found in the bottom of the reactor vessel. Apparently the UO_2 was removed by the ion exchange and filtration system.

Operation with a defective fuel element in the SADE test loop has also been described.[9] This hollow-tube type of fuel assembly, designated SH-2, experienced a high-temperature transient on May 8, 1959, and increased loop activity followed. Upon removal of the fuel element from the loop, it was found that the bellows, fitted at the top of the element to allow differential thermal expansion of the inner and outer cladding, had cracked transversely in one of the convolutions. Even though steam had entered the cladding during about one month of continuous exposure, the UO_2 pellets were in good condition. The rate of fission-gas release from the failed fuel-element bellows was estimated at 100 μc/sec. Noncondensable gases are vented from the loop condenser via a vent condenser and holdup tanks to the stack off-gas system. Loop condenser condensate is returned to the VBWR condensate system.[5] It appears, therefore, that most of the fission gas released from the test element would be vented to the plant stack, while fission-product solids carried by the superheated steam might find their way into the VBWR main stream. No operating difficulties resulting from this activity release were reported except for the problems encountered during two unscheduled shutdowns in May 1959. These problems are discussed later in this review under the heading "Incidents."

Operating Personnel Exposures

The radiation exposures of operating personnel from Jan. 30, 1959, to Jan. 31, 1960, have been reported.[1,2] This information was obtained from film badges that were developed monthly. It should be noted that all fuel handling and changing are done by shift supervisors and reactor operators. Table VI-1 shows that the actual exposures of these people were below the allowable exposure of 2500 mr in a six-month period. The maximum individual exposure during the one-year period was 930 mr or about one-fifth of the allowable exposure of 5 r/year.

Components

Control Rods. The first set of control rods employed in this reactor consisted of boron carbide plates enclosed in stainless steel. Six 12-in.-wide rods and one $5\frac{3}{4}$-in.-wide rod were used. The absorbing section of each rod was 43

Table VI-1 PERSONNEL EXPOSURES

	Exposure dose, mrem		
	Shift supervisors	Reactor operators	Turbine operators
Av. group exposure, Jan. 1 to June 30, 1959	280	240	180
Av. group exposure, July 1, 1959, to Dec. 31, 1959	125	416	85
Max. individual accumulated exposure, Jan. 1 to June 30, 1959	400	350	320
Max. individual accumulated exposure, July 1, 1959, to Dec. 31, 1959	240	580	260

in. long and was filled with boron carbide blocks 2.870 in. square and $\frac{1}{4}$ in. thick. The control-rod cladding was $\frac{1}{16}$-in.-thick stainless steel, and the resulting total blade thickness was $\frac{3}{8}$ in. Small holes were drilled through the side plates to provide an outlet for the helium generated in the boron carbide by neutron absorption and to relieve any steam evaporated from water leaking in contact with the rod poison; and, on Feb. 7, 1958, after several months of reactor operation, a routine rod calibration test disclosed that initial withdrawal of the central control rod decreased reactivity instead of increasing it. Further study showed that erosion of B_4C near the top of the absorber section had taken place, reducing rod worth a few tenths of a per cent reactivity. The reactor was shut down.[3]

Two different types of replacement control rods were built and subsequently utilized. One type consisted of a 0.225-in.-thick stainless-steel plate containing about 1 per cent boron enriched to 90 per cent B^{10} clad with $\frac{1}{16}$-in.-thick stainless steel. The blade width was 18 in. A second type of rod consisted of vibratory-compacted boron carbide powder sealed in stainless-steel tubes 0.210 in. in inside diameter and 0.250 in. in outside diameter. The tubes were arranged side by side to form blades 12 or 18 in. wide. A nontight stainless-steel cladding over the tubes provided the necessary rigidity. The reactivity control available from these various blade materials was about equal to that provided in the first set of control rods.[10] Both types of rods operated satisfactorily up to the time of the plant modification begun in December 1959.

Control-Rod Drives. The seven control-rod drives are mounted above the top reactor shield. Drive shafts passing through seals at the reactor vessel connect the control-rod drives to the control blades. The drives are patterned after those used on the BORAX reactors.[10] The actuating mechanism is a double-acting air cylinder with the piston attached to the rod-drive shaft. Air at 50-psi pressure, supplied from the instrument air system, is applied to the lower side of the cylinder to drive the shaft up until a stop on the shaft contacts a motor-driven stop. When the motor-driven stop is moved up or down, the stop on the rod-drive shaft follows it. To scram the reactor, air pressure is removed from the lower part of the cylinder and a higher pressure (200 psi) is applied to the upper part, driving the rod-drive shaft down to insert the control blades into the core. A balancing cylinder at the top of the rod-drive shaft is connected to the reactor vessel so that the cylinder and the motor-driven stop do not have to overcome reactor pressure but only the friction of the mechanisms and the seals. A review of this control-rod drive appeared in the June 1961 issue of *Nuclear Safety*.[11]

Changes in the control-rod system had been planned for the reactor modifications scheduled to begin in October 1959. As part of this program, it was decided to take motion pictures of the operation of the control rods before the modification. This was done late in September, and a study of the pictures showed that the scram time was greater than set forth in the license application. This discovery was made on September 23, and the reactor was immediately shut down. A number of tests on the control rods were then made to determine the causes that might slow down the scrams. At the same time, a study was made of the effects of the attainable scram speeds on the hazards evaluations. This study revealed that the attainable insertion time of 0.3 sec for one-fourth of rod travel did not significantly increase the consequences of the accidents previously evaluated on the basis of one-half rod travel in 0.3 sec. These times were essentially independent of initial rod position, and they included a waiting period of about 0.1 sec between the time the scram button was pushed and the time the rod began to move. On

October 3 and 12, permission was requested from AEC to continue reactor operations with the slower scram time. On October 16 AEC responded with a request for additional information.[2] Extensive tests followed,[2] including, among other things, a parametric study of the rod scram speed as a function of certain test variables. These variables included temperature, reactor pressure, rod-drive pneumatic pressures, and seal materials. Tests with and without the rod blade attached to the drives were conducted. Special instrumentation and timing devices were used to measure scram speed for full and partial rod insertion. During these tests the reactor was operated at 2 Mw(t) to obtain the desired temperatures and pressures. After these tests the following conclusions[3] were forwarded to AEC on November 11:

> The measurements obtained demonstrate that the differences in scram time between operating and unpressurized conditions are due entirely to the rod drive shaft sliding seals. No reliable data are available to indicate changes in speed with time, but they appear to be small. None of the seals which have been removed appear to have changed as a result of aging. One can infer from these results that the control rods at the time of their original installation in the vessel operated in about the way they do now. That the performance of the rods is consistent and reproducible is apparent from the intercomparison of the many tests made during the course of this investigation. Typical variations in time for successive tests on the same rod were on the order of 10%. Differences between rods were found to be due to mechanical alignment, and it was possible to correct this so that the times required for the rod to travel one-fourth the stroke were uniform to plus or minus 15% or less.

On Dec. 2, 1959, permission was requested from AEC to conduct further reactor tests for determining the effects of some of the changes that were planned to be made in the course of the approaching reactor modification.

By the middle of December 1959, permission to resume operation had not been received from AEC, and the previously scheduled modification of the reactor was begun. (Reactor operation after the modification was approved on the basis of an insertion time of 0.3 sec for one-third of rod travel. This, however, applied to a different core and a different set of control blades.)

To clarify any future difficulties with the drive-shaft sliding seals, it was intended to test initially, at operating temperature and pressure, at least one pair of the control rods sufficient times to obtain reliable statistics on drop time.[12] The remaining rods were to be tested a sufficient number of times to show that all rods behaved in a similar manner.

In addition, a separate pneumatic system was to be provided for each set of two control-rod drives. All conventional lubricating oil was to be eliminated from these air systems to minimize the explosion hazard.[12]

Reactor Vessel Liner. The modification program begun in December 1959 required removal of the reactor vessel head, and thus there was an opportunity to inspect the reactor vessel liner.[4] This liner, made of type 304L stainless steel, was roll bonded to the vessel and the head. It was seam and plug welded to both the head flange and the vessel flange. Cracking of the vessel flange liner was found at both the seam and plug welds. There was no cracking of material remote from the welds. Although the plug welds had appeared to be satisfactory, most of them had not penetrated sufficiently into the flange. Early failure of these plug welds led to excessive stressing of the remaining seam and plug welds during vessel thermal cycling. The vessel had been heated to 550°F and cooled to room temperature about 200 times. After removal of the vessel liner, all the welds were ground down and dye-penetrant inspected. No cracks had penetrated into the vessel flange. At the time Kornblith and Leyse reported on the VBWR,[4] the head flange had not been inspected, but visible evidence indicated a similar failure of the head flange liner. It was anticipated that a replacement carbon steel liner would be installed on the head flange and an Inconel liner on the vessel flange. Attachment was to be by welds at the edges only, eliminating the plug welds. Thermal stresses in the welds should be reduced, since the liners and the flanges they are attached to will have comparable coefficients of thermal expansion in the encountered temperature range.

Other Failed Components. The subsequent section of this review on "Incidents" describes failures of the main steam-pressure-reducing valve, failure of the gasoline-driven compressor supplying air for emergency air cooling in the SADE loop, and failure of some of the SADE loop instrumentation power. Detailed accounts of the prime causes of these failures have not been published. An accumulator was added to the SADE loop cooling-air system in an attempt to

render emergency loop cooling independent of electric power or the operation of a gasoline engine.

Reactor Kinetics

The initial core for this reactor consisted of 3-in.-square fuel elements[10] fabricated from thin stainless-steel plates containing fully enriched UO_2 clad with stainless steel. In addition, single experimental fuel elements of promising design were inserted into regular core positions from time to time.[3] Initial power operation was with an approximately 3-ft-diameter core of 101 fuel elements, containing a total of about 22 kg of U^{235}. The temperature coefficient was slightly positive at 70°F (0.6×10^{-5}/°F) and decreased linearly with temperature.[7] Above 106°F the temperature coefficient was negative. The void coefficient (Δk times the ratio of water volume to void volume) was negative,[7] −0.04 at 70°F and −0.1 at 545°F. The reactor was operated at all powers up to 30 Mw(t), with pressures between 900 and 1000 psi, smoothly and with no sign of divergent oscillation.[3] It was also operated for short periods of time at various power levels at 600 psi with similar results. Under steady-state conditions, the operation of the control rods consisted of movement of one rod about 0.1 per cent of its travel two or three times per hour. The reactor responded very smoothly to changes in turbine output, and almost instantaneous load changes of 25 per cent of full load could be made with a reactor pressure fluctuation of only 10 psi. The turbine load could be increased from zero to full power in less than 15 sec without severe drop in reactor pressure. These changes were all made without bypassing any steam to the condenser.[3]

On Jan. 30, 1959, authorization was given for operation with an entire core of low-enrichment, rod type fuel elements in the manner described in the Final Hazards Summary Report.[10] These fuel elements consisted of 2.3 per cent (uranium element basis) UO_2 pellets clad with either Zircaloy or stainless-steel tubes. Both rod type elements were similar in construction and materials to those to be used in the Dresden reactor. Operation with cores containing mixed types of fuel elements was also carried out. Initial operation with new types of fuel involved only one assembly at a time. Special procedures were followed to carry out safe, controlled tests of these elements. Proceeding with the next step in these tests was contingent upon successful completion of the previous step. Between Jan. 30 and July 31, 1959, a series of tests,[1,3] was made as part of a program established in the license applications in order to show that cores consisting of approximately equal numbers of two very different types of fuel could be operated safely. In no case was there any indication of oscillations. The peak-to-peak values of noise signals were on the order of 6 per cent of the steady-state signals at a power of about 27 Mw(t) and generally less at lower powers. Attempts were made to ascertain the periodicities in the noise by means of autocorrelations and Fourier analysis. The results showed that there were no large resonances present, but the analysis was not sufficiently refined to allow separation of any small resonances which might exist from the random noise.

Power operations were carried out principally under conditions of single-cycle natural circulation, although dual-cycle and forced-circulation modes were tested for short periods.

Incidents

Several incidents that are of interest took place during this report period. On Mar. 9, 1959, the valve plug separated from the stem in a 6-in. pressure-reducing valve. The purpose of this valve was to reduce the reactor steam pressure from a nominal 1000 to 450 psi, the turbine design pressure. This failure almost instantaneously stopped the flow of steam while the reactor was operating at 30 Mw(t). The pressure in the reactor rose following the sudden decrease in heat removal, and as a result the core void fraction decreased. This latter event introduced positive reactivity and raised the reactor power to 35 Mw(t), the setpoint for automatic high-flux scram. The reactor scrammed about 5 to 10 sec after the valve failure. The reactor pressure had risen by about 50 psi at that time.[1] It was not possible to tell from the recorder whether the pressure continued to rise in the first few seconds after the scram, but, if so, it was included in the pressure rise cited above. A direct comparison with the hypothetical loss-of-steam-flow accident discussed in the final hazard report[10] is difficult because of differences in core configuration and in reactor power preceding the incident. For a core with the rod type fuel elements, the

hazard report[10] predicted a high-flux scram at 125 per cent power [25 Mw(t)] 4 sec after the loss of flow. Reactor pressure was calculated to have risen 25 psi at this instant. The equivalent numbers for the core with all plate type fuel elements, taken from the hazard report,[10] are high-flux scram at 125 per cent power [25 Mw(t)] about 10 sec after the loss of flow. Reactor pressure in this instance was estimated to have risen by about 150 psi. The actual core on March 9 consisted of about 25 per cent rod elements and 75 per cent plate elements. It appears then that the scram occurred within the time limits predicted and that the rate of pressure increase may have been less than expected.[1] No damage resulted from this accident. The emergency heat exchanger (1200-kw capacity) would have been available during this incident, since it is connected to the reactor steam system upstream of the failed pressure-reducing valve. However, system pressure did not rise enough to call for its operation.

On May 8, 1959, the reactor scrammed automatically as a result of loss of the electric power supply to the VBWR site. The reactor was being brought to power at that time and was operating at slightly under 5 Mw(t) when the outage occurred. The reactor-building isolation valves in the ventilating system had been designed to "fail safe" and consequently closed upon the loss of power. Five minutes after the power outage, a gasoline-driven air compressor providing emergency air cooling of the superheat loop failed due to ignition trouble.[3] The shift from steam cooling to emergency air cooling could consequently not be made, and the superheat test-element cladding temperature rose to an estimated 1500°F, 300°F beyond the design operating limit. The loop exit temperature could not be measured during this transient because the a-c power supply had also failed. After the shutdown a rise in loop activity was observed.[9] As mentioned before, the ventilation-system isolation valves had closed upon the loss of power. This allowed activity from the superheat loop, escaping via several small steam leaks, to accumulate in the reactor building. Subsequently, the building was evacuated for several hours.[3]

On May 10, 1959, after reactor and superheat loop operation had been resumed, the reactor building high-pressure switch was accidentally bumped and the reactor safety system was actuated. The reactor was scrammed automatically, and the ventilation-system isolation valves were closed automatically. Contaminated steam from the superheat loop was still leaking into the reactor building at this time, and it caused an increase in the building air activity when the ventilation system was secured. It appears that the reactor building was then again evacuated for a short time. In neither of these two incidents was there any excessive exposure of personnel or damage to property.[3]

Subsequently, several changes were made to the superheat loop in an attempt to provide more reliable emergency cooling to the loop in the event of power failure.[5] The first change involved the addition of an accumulator in the cooling-air supply. Thus, if a power failure occurred during startup when the experiment was normally air cooled, the air accumulator would automatically provide cooling air. The accumulator is sized so that, by the time the air supply is exhausted, the decay heat production in the element will have been reduced to the extent that the design cladding temperature of 1200°F will not be exceeded.

At the same time, the steam bypass valves, in parallel with the loop flow control valves, were provided with an emergency d-c power supply. If, during steam cooling, the loop flow control valves should close because of a loss of power or operator action, the bypass valves will automatically stay open for 15 min. After this delay time, cooling of the test element by radiation and convection of heat directly through the process tube wall will be adequate. After the 15-min delay time, steam flow through the bypass valves can be controlled by the operator.

Several conclusions may be drawn from these incidents. One is that the consequences of a plant failure may be seriously aggravated if the plant is operated with a partially violated line of defense against the spread of radioactivity. In this instance small leaks in the loop steam system contributed to partial voiding of the containment provided by the primary system. Although excessive personnel exposures to radioactivity were avoided in the two separate incidents, the fact remains that the reactor building was evacuated because of faults that in themselves did not release any activity. Evacuation of a reactor plant during a power outage seems particularly inopportune, since manual operator action may be required at this time to place the plant in a safe condition.

Another conclusion—one that is not new—is that emergency electric generators may not operate when needed in an emergency. The air accumulator added to the loop emergency cooling system should have much greater functional reliability than a gasoline-powered generator, provided a full charge of air can be assured. In these particular instances operator action within the containment building would have been feasible by men wearing fresh air masks, since the direct radiation level was not excessive.

Furthermore, it is concluded that great care must be taken in providing a reliable power supply for instrumentation. Although measurement of the superheater loop exit temperature was part of the reactor safety system, failure of its power supply did not prevent a reactor scram due to loss of power. However, it still appears desirable that information should be available to the operator during a loss-of-power accident so that he may be able to take all action available to him to minimize the consequences of this failure.

Failure of the main steam-pressure-reducing valve demonstrated that the commonly postulated step flow changes in reactor transient analysis may be quite realistic after all. It is encouraging to note that, in spite of the uncertainties and assumptions that are inherent in a loss-of-coolant-flow transient analysis, actual response of the reactor fell within the range of values calculated prior to the incident.

There were several less serious accidents involving release of radioactivity.[3] Two incidents that occurred in November 1957 and on Jan. 18, 1958, respectively, involved the release of some water from the reactor system into the drain system following operator error. The amount of activity released is stated to have been trivial. On Nov. 10, 1958, a sticking valve in a makeup water system caused the release of about 200 μc of beta and gamma emitters to a pond on the site. The pond area was several acres, and it is stated that the concentration of activity was many orders of magnitude below the maximum permissible levels. On Feb. 1, 1958, the reactor building was evacuated temporarily when several tiny radioactive particles were found on the building floor. The contamination resulted from the cutting of wire that had been inserted in the reactor for measuring neutron fluxes. None of the particles were scattered outside the reactor building. On Feb. 4, 1958, a technician became slightly contaminated by water dripping from a leak in a sight glass. The contamination was easily removed, and it did not spread beyond the reactor enclosure. On Apr. 27, 1959, a spread of contamination resulted when contaminated water spilled from a transfer cart onto the ground and paved area near the reactor building during the transfer of an experimental fuel element from the reactor building to the Radioactive Materials Laboratory. The contamination was quickly discovered and successfully removed.[3] Property damage was less than $1000, and the loss of time of the reactor facility was less than one day.[1] In none of these activity releases was there any excessive exposure of personnel, nor was any excessive amount of activity allowed to leave the site through any of the effluent systems.[3]

This last group of incidents demonstrates again that, even with great care in the selection of equipment and in the training of personnel, complete control of activity is extremely difficult. There does appear, however, to have been a reduction in the frequency of incidents due to operator error as operating experience was gained with the reactor. (O. H. Klepper)

Heavy-Water Leaks at Savannah River

Three independent equipment failures occurred in the Savannah River reactor plants during 1960 and resulted in some leakage of heavy water. First, there was a reactor outlet nozzle leak.[13,14] The exact cause of the leak is undetermined, but presumably it was related to extensive intergranular attack that was probably caused by pickling solutions. Next, there was a fatigue failure in a drain line.[14,15] This failure was probably caused by vibrations in the piping. Then there were heat-exchanger tube failures[13,14] caused by the stress-corrosion cracking induced by chlorides on the exterior surfaces of tubes exposed to the atmosphere.

All these failures were discovered in the course of normal inspections, and the equipment involved has been repaired or replaced. Similar conditions in all duplicate installations have been checked, and all practical measures have been taken to guard against other failures of the same type. The failures did not result in any radiation incidents or misoperation of the reactor, but each carried the potential hazard

of loss of moderator from the reactor system, flooding of the circulating pumps, and resultant failure of the reactor cooling systems. These incidents, however, and particularly that involving the reactor nozzle leak, are of concern in that they constitute a mild version of the loss-of-coolant incident, extreme instances of which in most power-reactor installations constitute the maximum credible accident.

The Savannah River Plant reactors are enclosed in vertical, cylindrical, stainless-steel tanks in which the fuel and the materials to be irradiated are suspended. Heavy water is circulated through the tanks to serve as moderator and coolant and is cooled by river water, which is pumped through the shell side of the heat exchangers. The heavy water is extremely pure; it contains less than 6 ppb chloride and is maintained at a *p*H of 5.7 by the addition of nitric acid. The heavy-water system is constructed of AISI types 304 and 304L stainless steel.

Extreme care was taken in construction to obtain the most reliable system that was practical. In particular, the heat exchangers had been rigorously inspected and tested before acceptance. Specifications were provided for all fabrication details, including welding, heat-treating, radiographing, tube rolling, Freon and hydrostatic testing, and cleaning. Specifications to ensure cleanliness of those parts in contact with the moderator were particularly stringent. Various methods were used throughout the fabrication procedure to ensure freedom from atmospheric dirt, grease, and free iron.

The secondary cooling water is Savannah River water that is untreated, except for intermittent addition of chlorine for control of algae. The chloride content of the untreated river water is normally within the range 2 to 9 ppm.

Reactor Outlet Nozzle Failure

Heavy water flows through the reactor from top to bottom and leaves the tank through six discharge nozzles spaced equally around the bottom. The nozzle cross section varies from approximately rectangular at the tank wall to circular at the end of a 90° bend.

Specimens of the nozzle that failed were examined[13] by visual, macroscopic, microscopic, radiographic, dye-penetrant, physical, and weld testing techniques. The results of these examinations indicate that pickling may have damaged the surface (intergranularly) of the nozzle wall to a depth of as much as 0.040 in. and that initiation of cracking was due to a field weld made about 1 in. from the damaged area during installation of the reactor. Repair necessitated excavation of the shield, and the high radiation levels encountered required temporary shielding to limit personnel exposures.

Drain Line Failure

During an inspection on Apr. 7, 1960, a leak was discovered at a 2-in. branch connection on a 6-in. reactor drain line. The leak rate was 1 to 2 drops per minute at a location that could not be isolated from the reactor tank for repair. The leakage rate increased to 50 drops per minute when collar bands designed to support tie rods between the 6- and the 2-in. lines were tightened on the 6-in. lines. This experience indicated that the leak might increase if disturbed. Because complete failure would have resulted in a leak rate of several hundred grams per minute, the fuel was discharged, the reactor was drained, and the failed section was removed.

Liquid penetrant examination of the outer surface of the removed section of 6-in. pipe indicated a crack extending around approximately one-fourth of the circumference of the weld joining the 2-in. connection to the 6-in. pipe at the edge of the weld.

Metallurgical examination showed this to be a fatigue failure that was probably caused by vibration in the piping. No cracks were found in other similar locations.

Heat-Exchanger Tube Leaks

The heat-exchanger tubes are ½-in.-OD, 18-BWG AISI type 304 stainless steel rolled into double tube sheets at both ends and welded to the outer tube sheet at each end. The tube sheets and the shell are of AISI type 304L stainless steel. The shell is welded to the inner tube sheets, and the spaces between the inner and outer tube sheets are closed by carbon steel rings that slide into position against the outer tube sheet and are sealed by elastomer gaskets.

The heat-exchanger tubes failed in the leak-collection space between the double tube sheets. In one exchanger most of the 8957 tubes were found to contain cracks, although the leak rate was only about 2 gal/hr when the exchanger was removed from service.

Samples of deposits (principally Fe_2O_3) removed from tubes in the heat exchangers and samples of the gaskets used in the leak-collection space were analyzed for water-soluble chlorides. The deposits removed from one heat exchanger that failed contained 300 ppm chloride; the gasket from this exchanger contained 8 per cent chlorine. Analyses of other exchangers showed the chloride content of deposits to be between 100 and 4000 ppm and the chlorine content of gasket materials to be from 0.4 to 30 per cent.

Examination of selected tube sections from the two faulty heat exchangers revealed that tube sections from the moderator inlet leak-collection space had typical transgranular stress-corrosion cracks. Tubes in the most severely damaged heat exchanger were cracked throughout the tube bundle in the leak-collection space at the moderator inlet (hot) end. No stress-corrosion cracking was found on tube sections removed from the leak-collection spaces at the moderator exit (cold) end or from inside the cooling-water shell of either exchanger. Cracking did extend into the rolled area of the tube-to-outer-tube sheet joint. Cracking did not extend into the rolled area of the tube-to-inner-tube sheet joint. It was apparent that the cracking originated on the outer surfaces of the tubes. Flattening tests of some tube sections indicated severe cracking, and the outer surfaces of some tubes were heavily pitted.

As a precautionary measure, all the heat exchangers are being inspected, and a corrective program of modifications is under way. Two exchangers with leaks of about 2 gal/hr were replaced, three in service are leaking (October 1960) from 0.3 to 4.0 gal/day, and several others show slight leakage.

Laboratory and field corrosion tests are in progress to determine suitable chemical and galvanic inhibitors to prevent initiation of stress-corrosion cracking and to prevent crack propagation where cracking has already begun. Once a suitable inhibitor is available, corrosion inhibition will be accomplished by flooding the leak-collection space with the appropriate inhibitor. This flooding will also prevent alternate wetting and drying of the tubes and resultant concentration of chlorides. The leak-collection spaces of several heat exchangers have been flooded with an alkaline-water solution containing a phosphate inhibitor and a sulfite scavenger for oxygen to prevent corrosion of the carbon steel seal ring.

Conclusions

These failures are of interest because they occurred in carefully constructed stainless-steel systems after several years of operation at relatively low temperatures. The operating record is very good in that the failures represent a very small part of all the components built and operated at Savannah River. Since these difficulties were detected and corrected in an early stage of development before serious consequences resulted, the effects of the failures were minimized and the value of routine inspection procedures during operation was demonstrated. The double tube sheet proved its value in detecting leaks and in keeping the two fluids separate, but, in the case described here, the failure was a direct result of the presence of the space between the inner and outer tube sheets.

Only one of the failures described occurred in a welded joint. All the others were in base metal, formed plate, or tube walls. Great care is expended in fabrication and inspection of welded joints in this type of construction, with relatively little attention given to soundness and surface conditions in base metal outside the immediate vicinity of the welds. Thus it is possible for defects resulting from overpickling or other causes to exist outside of weld areas without being discovered.

It is not likely that all causes of possible future failure can be detected at the time of construction. Therefore it cannot be assumed that a vessel or pipe buried in an inaccessible part of a concrete shield structure will never need to be repaired or removed. Certainly the outlet nozzle failure indicates the desirability of having all parts of the system removable or accessible for repair wherever practical.

The susceptibility of stainless steel to intergranular and stress corrosion under certain conditions has only become well known in recent years. At the time of the design and construction of the Savannah River Plant, the conditions contributing to stress corrosion were not so well known. There is no evidence that the failures were associated with or accelerated by nuclear radiation or radioactive contamination. The damage found and the circumstances sur-

rounding the failures could occur in nonnuclear industrial practice. (W. R. Gall)

Source Incident at Nevada Test Site

A technical supervisor and a technician received gamma-radiation doses of about 5 and 18 rem, respectively, as determined by film-badge dosimetry, during routine calibration of photodiode detectors at Station 311 of the Nevada Test Site (NTS).[16] The maximum permissible doses are 3 rem per 13 weeks and 5 rem per year. Although the exposures in this incident are not particularly serious, the incident is reviewed here as an example of what may be the most probable and frequent type of radiation incident: the accidental or negligent misuse of radioactive sources.[17,18]

The two men were calibrating detectors in the radiation field of a 340-curie Co^{60} source on Oct. 4, 1960, when the exposures occurred. It seems most likely that the cause of the personal dosages was that both men walked into the beam from the source one or more times while accomplishing the calibrations without realizing that the beam port was open. The exposures were discovered on Nov. 28, 1960, when film badges worn by the men were routinely processed.

The investigating board found[16] that the two men did not follow the established procedure for the operation and that there was no equipment malfunction. It found that the operating procedures were not specific enough; there was a delay by the men in turning in their film badges; there was a lack of followup by management (although some immediate corrective measures were taken); and inadequate use was made of available monitoring equipment. The board made the following specific recommendations[16] for preventing recurrence of such incidents:

1. Operating procedures should be reviewed and revised to reduce as far as possible the chance of overexposure of personnel. This should be accomplished in part through specific assignment of the responsibility for monitoring during this type of operation.
2. Film badges should be exchanged upon completion of any work involving radioactive sources.
3. All personnel working in radiation areas should have, on their person, an instrument which would immediately sound an alarm indicating that radiation is present. . . .
4. A physical barrier, with an interlocking device, should be provided for operation of this source, and any similar sources.
5. All personnel should be thoroughly and periodically schooled in proper radiation safety procedures for this type of operation.

In addition, the investigating board made the following general recommendations[16] for future operations at NTS:

1. All agencies involved in radiation work at NTS should coordinate their operations with the NTS Rad-Safe organization.
2. Nevada Test Site Rad-Safe should be required to review all radiological safety operations at NTS periodically and these findings brought to the attention of the Las Vegas Area Operations for necessary action.
3. The entire procedure of monitoring by film badges, dosimeters, and other devices should be reviewed periodically by a committee established by the Area Manager, Las Vegas Area Operations. This committee will determine whether established criteria and personnel monitoring methods adequately comply with AEC regulations.
4. Rad-Safe monitors present at an operation should be briefed in the operation to the extent that such monitor or monitors are virtually an operational member of the team.

(L. A. Mann)

AEC Actions on Regulations

Rules of Practice

The Commission published[19] on Apr. 15, 1961, an amendment to Part 2, Title 10, *Code of Federal Regulations*, to establish an immediate review procedure for licensing and compliance proceedings. The amendment was to be effective upon publication. The amendment provides that an intermediate decision of the hearing examiner shall become final within 20 days after the decision, unless a petition for review is filed and the Commission grants the petition. Granting of review petitions will ordinarily be limited to considerations of applicable statutes and rules and important or novel questions of law or policy. The Commission reserves the right to review any case and to make the final decision. This step toward lessening the number of proceedings required and toward expediting and limiting Commission reviews is believed by the Commission to be in the public interest.

Licensing of Source Materials

An amendment, effective Apr. 11, 1961, was made to Part 40, Title 10, *Code of Federal Regulations*, to reestablish exemption from licensing for "rare earth metals and compounds, mixtures and products not containing more than 0.25% by weight thorium, uranium, or any combination of these."[20] This action was based on a Commission finding that the national security was not involved and that there was no reasonable hazard to life or property.

Licensing of Production and Utilization Facilities

Changes Without Specific Authorization. Proposed amendments to Part 50, Title 10, of the *Code of Federal Regulations* were published on May 9, 1961, for review and comments. The proposed amendments concern changes that licensees can make in facility design and operation without specific AEC authorization in the licenses.[21] These amendments are based on the Commission's "Memorandum and Order" of Nov. 2, 1960, concerning the license for the Vallecitos Boiling Water Reactor (VBWR) at Pleasanton, Calif.

The amendments provide that:

1. Certain significant design and operating limitations are to be designated as technical specifications and must be adhered to in the absence of specific authorization from the Commission.

2. The licensee may make changes in the facility, make changes in procedures described in the hazards summary report, and conduct tests or experiments, unless the proposed change, test or experiment involves a change in technical specifications or an unreviewed safety question. Such changes, tests, and experiments must be reported promptly to the Commission. An unreviewed safety question involves a possible increase in the probability or consequence of any accident previously analyzed in the hazards summary report or a credible probability that a different type of nuclear accident could occur than was previously considered and reviewed.

3. If a proposed change, test, or experiment involves a change in the technical specifications or an unreviewed safety question, it may not be carried out unless authorized by the Commission. The request for such authorization must include an appropriate hazards analysis.

4. Except as provided in paragraph 5 below, the Commission may authorize any proposed change, test, or experiment upon finding that there is reasonable assurance that the health and safety of the public will not be endangered.

5. With respect to power and testing reactors, the Commission may authorize a proposed change, test, or experiment if it finds that this does not involve significant new hazards considerations and that public health and safety would not be endangered. If the proposed change involves significant new hazards considerations, the question must be referred to the Advisory Committee on Reactor Safeguards (ACRS) for review, and a public hearing must be held.

As a guide to licensees and applicants, an appendix to the proposed amendments to Part 50, Title 10, of the Code lists the items to be covered by technical specifications for each kind of facility. A period of 60 days was to be allowed for the submission of comments.

Mortgages and Liens Facilities. The Commission published for review and comments, on May 9, 1961, proposed amendments to Part 50 which would grant general approval to the mortgaging of reactor facilities.[22] In the interest of safety, the same license requirements would apply to the creditor as apply to the licensee, and no creditor could take possession of a facility without a license from the Commission authorizing the transfer. The Commission had previously consented to mortgages on the Yankee and Dresden facilities following public hearings. As mentioned above, a period of 60 days was to be allowed for comments and suggestions regarding the proposed amendments.

Financial Protection Requirements and Indemnity Agreements

On Mar. 31, 1961, the AEC forwarded its annual report on operations to the Congressional Joint Committee on Atomic Energy (JCAE) as provided in Section 170 of the Atomic Energy Act of 1954 (as amended). The indemnification of atomic energy facilities and the activities of the ACRS are included in the report.[23,24] Some interesting features of the report are mentioned below.

An amendment[24] that was published in the *Federal Register* on Feb. 17, 1961, increased the population factor for calculating financial protection requirements from a minimum of 1.5

to a maximum of 2.0. No change was made in the "base amount" of the formula for protection, $150/kw(t), or in the minimums required of relatively low power levels. The Commission concluded that the maximum population factor of 4.0 recommended by the insurance syndicates would be too high and that a maximum of 2.0 was reasonable.[26]

In March 1961 the Commission adopted two effective and two proposed amendments to Part 140 of the Regulations. The two effective amendments establish forms of indemnity agreements, with licensees furnishing financial protection, to cover the "common occurrence gap" in coverage, the time gap between occurrence of an incident and reestablishment of protection, and other previously cloudy questions.[23] The two proposed amendments would establish forms of agreements with federal agencies and nonprofit educational institutions not required to furnish financial protection.

On April 8 the Commission recommended legislation to eliminate indemnity coverage of "on-site" property based on the philosophy that the purpose of government indemnity is to protect the public, rather than to protect the facility owner from on-site property damage, and that such coverage reduces the total coverage of the public.[23]

The Commission invited comments on the question of whether processors of unirradiated uranium enriched in U^{235} (not U^{233} or Pu^{239}) should be indemnified and required to furnish financial protection.[23] Partially conflicting comments were received from the Atomic Industrial Forum and from the insurance syndicates.

The Commission and the insurance pools reported that no claims had been received as of Feb. 28, 1961, under the licensee insurance and indemnity agreements. Also, no claims have been filed with the Commission under its indemnity coverage of contractors.[23]

The Commission reported progress on maritime indemnity problems and discussed problems of foreign indemnity, stating that, "Developments within the past year have substantially advanced the United States objective of attaining adequate international standards governing civil liability for nuclear hazards."[23]

Amendments to Part 140 were proposed on Apr. 14, 1961, by the Commission to allow different and clearer wording in insurance policies.[27] The amendments were effective after 30 days.

Another amendment[28] was added on Apr. 22, 1961, to be effective after 90 days. This amendment removes the "common occurrence" gap (defined in the insurance policies) in financial protective coverage by permitting indemnity to be applicable at the level at which insurance coverage would terminate in such a case.

(L. A. Mann)

AEC Separation of Regulatory Functions from Operation and Development Functions

Study by the Professional Staff of the JCAE

The professional staff of the JCAE in March 1961, following a study of the separation of regulatory and promotional activities within the AEC, recommended[29] establishment of a three-man Atomic Safety and Licensing Board within the AEC. Board members would be appointed by the President and confirmed by the Senate. The Board would have final licensing authority and would recommend regulatory standards and rules to the Commission, which would retain the responsibility and authority for making rules and setting standards. The study spelled out details of the proposed separation of specific authorities and responsibilities recommended with the aim of alleviating the problems engendered by the mixture of promotional and regulatory functions within the Commission. It was pointed out that the report[29] of the study was not a JCAE report and that it did not present the views of the JCAE or its individual members.

AEC Announcements of Actions

Chairman Glenn T. Seaborg of the AEC announced on Mar. 16, 1961, that the Commission had acted to separate its regulatory functions from its operational and development functions at the General Manager level.[30] Pending completion of the staffing report, the Division of Licensing and Regulation, the Division of Compliance, and the Office of Health and Safety were transferred to the Office of Director of Regulation. Harold L. Price was named the Acting Director of Regulation to report directly to the Commission. General Manager A. R. Luedecke was named to continue as administrator for operational and developmental activities.

In the February hearings and in compliance with Section 202 of the Atomic Energy Act, AEC Commissioner Loren K. Olsen had indicated to the JCAE[31] that the Commission believed such a separation of responsibility would be both advantageous and a natural development of modifications previously made. The above-mentioned change was the first step in the plan.

On Apr. 25, 1961, following completion of the staffing study, the Office of Health and Safety was abolished. A new Office of Radiation Standards, whose director is Forrest Western, was created by Harold L. Price, and a new Office of Operational Safety, whose director is Nathan H. Woodruff, was created under the General Manager.

On Apr. 28, 1961, Deputy General Manager Hollingsworth announced[32] that "Effective May 1, 1961, the licensee inspection and investigative functions and responsibilities assigned by AEC 0704, 0705, and 0707 to Manager of Operations and Heads of Divisions and Offices, Headquarters, are transferred to the Director, Division of Compliance, Headquarters."

On May 10, Robert Lowenstein, Acting Director of the Division of Licensing and Regulation, announced[33] organizational changes in that Division which became effective as of Apr. 17, 1961. The staff members and their titles designated in the announcement were:

Division Director Staff

Acting Director . . . . Robert Lowenstein
Deputy Director . . . . Richard L. Kirk
Assistant Director . . . . Eber R. Price

(a) Reactor Standards Staff . . . . . Joseph J. DiNunno, Chief

Responsible for developing safety standards, criteria and guides for the location, design, construction and operation of nuclear reactors and of major components and assemblies of such reactors.

(b) Enforcement Branch . . . . . Richard Cunningham, Acting Chief

Reviews reports of compliance inspection of reactor licensees and of source, special nuclear and byproduct material licensees and takes appropriate action thereon including recommendations for enforcement actions. Participates in the investigation of incidents involving the licensed operation of reactors, use of source, special nuclear, and byproduct materials, and recommends appropriate action with respect thereto for protection of radiation workers and the public.

(c) Administrative Branch . . . . . Chief (to be designated)

Develops and recommends policies and procedures for the indemnification of licensees against public liability arising out of nuclear incidents; provides general administrative services to the Division; preparation of Division reports; and develops procedural manuals and instructions for the internal operations of the Divisions.

Assistant Director for Facilities Licensing . . . Dr. Clifford K. Beck

(a) Research and Power Reactor Safety Branch . Edson G. Case, Chief

The Research and Power Reactor Safety Branch is responsible for hazards evaluation of the design, construction and operation of the following types of reactors and of other types of production and utilization facilities as assigned; and for processing license applications for such facilities:

Boiling Water	Organic Moderated
Pressurized Water	Research Training
Graphite Production	Critical Assemblies

(b) Test and Power Reactor Safety Branch . . Dr. Martin Biles, Chief

The Test and Power Reactor Safety Branch is responsible for hazards evaluation of the design, construction and operation of the following types of reactors and of other types of production and utilization facilities, as assigned; and for processing license applications for such facilities:

Test Reactors	Gas-cooled
Sodium Cooled	Heavy Water Production

This Branch is also responsible for developing and administering examinations to applicants for reactor operators' licenses.

(c) Criticality Evaluation Branch . . . . . Dr. Charles Luke, Chief

Responsible for the technical review and analysis of applications for facilities and materials licenses and for recommending appropriate action thereon with respect to the adequacy of design, process, and operating procedures to assure against accidental criticality in the processing and storage of special nuclear materials. Performs similar reviews and makes appropriate recommendations with respect to license applications involving the transportation of special nuclear material and of irradiated fuel elements.

(d) Site-Environmental Branch . . . . . John Newell, Chief

Responsible for the safety evaluation of proposed nuclear facility sites against stand-

ards and criteria relating thereto, including meteorological, hydrological and seismological factors, and waste disposal; and liaison with Federal and State Agencies in these areas.

(e) Special Projects Branch . . . . . Chief (to be designated)
Responsible for the investigation and analysis of specialized reactor safety problems; identification of areas requiring research in reactor safety; liaison with the Nuclear Safety Journal and other technical publications to assure timely circulation of reactor safety information to the scientific and industrial community and to members of the AEC technical staff; handling other specialized reactor safety problems not falling within the assignments of other Branches reporting to the Assistant Director for Facilities Licensing; and safety reviews of nuclear applications in aerospace activities.

Assistant Director for Materials Licensing . . . Lyall Johnson

(a) Source and Special Nuclear Materials Branch . . . . . Donald Nussbaumer, Acting Chief
Reviews, evaluates and processes to final action all applications for licenses to receive, possess, use, and transfer source and special nuclear materials and allocates such materials to licensees. Recommends policies and procedures for the regulation, licensing and distribution of source and special nuclear materials. Forecasts requirements and maintains allocation records for special nuclear materials distributed pursuant to Section 53 of the Atomic Energy Act.

(b) Isotopes Branch . . James R. Mason, Chief
Develops, recommends, and administers policies, criteria and procedures for the regulation and licensing of byproduct materials. Performs technical evaluations of radiological protection problems associated with applications for byproduct material licenses. Issues, amends, renews, or denies licenses to receive, possess, use and distribute byproduct materials.

Assistant Director for Materials Standards . . . Lester R. Rogers
Develops safety standards, criteria and guides to assure protection of the public and radiation workers in the licensed use of byproduct, source and special nuclear materials.

On May 12, 1961, Chairman Seaborg announced the following additional staff changes approved by the Commission in its regulatory program:[34]

(1) The name of the Office of Health and Safety has been changed to the Office of Radiation Standards. Dr. Forrest Western will be Director to this office, which will perform the functions of a regulatory nature handled in the past by the Office of Health and Safety. These functions are the development of basic radiation standards, including staff assistance in connection with the work of the Federal Radiation Council, and the development of the AEC-State relations program in the field of licensing and regulation, including work being done towards the transfer of certain AEC regulatory responsibilities to the states.

The remaining functions of the former Office of Health and Safety, including research, industrial safety and radiological assistance, are combined in a new Office of Operational Safety reporting to the Commission's General Manager, A. R. Luedecke. The safety research function will include not only the stimulation of research but also the encouragement of the atomic energy industry to use the results of this research. Dr. Nathan H. Woodruff will be Director of the Office of Operational Safety.

The General Manager retains the management safety responsibilities in carrying out AEC operations. He is responsible for assuring that AEC operations are planned and carried out in such a way as to protect the health and safety of Government and contractor personnel and the general public. AEC operations will be subject to such review by the regulatory staff as the Commission may direct. The General Manager also is responsible to the Commission for protecting Government property.

(2) Field inspection personnel connected with the regulatory program in AEC Operations Offices have been transferred to the direct authority of the Director of the Division of Compliance, L. D. Low. Previously, these inspection personnel have reported to the manager of the Commission's Operations Offices.

(3) Some changes in organization have been made in the Commission's Division of Licensing and Regulation to divide into separate components the responsibilities for licensing, for enforcement proceedings, and for development of guides and regulations. For example, a separate reactor standards staff has been established within the Division of Licensing and Regulation to focus increased attention on the development of regulations and guides for siting and operation of nuclear reactors.

Under the revised organization of the Division of Licensing and Regulation, all work connected with reactor licensing, including safety evaluation by the technical staff and administrative work connected with the issuing of licenses and amendments, now is under the direction of Dr. Clifford K. Beck, Assistant Director for Facilities Licensing.

Lyall E. Johnson, Assistant Director for Materials Licensing, now is responsible for the Source and Special Nuclear Materials Branch and the Iso-

tope Branch, which handle the issuance of licenses for those materials. Lester R. Rogers, as Assistant Director for Materials Standards, will be concerned with developing standards for the possession, use and shipment of radioactive material.

A separate enforcement branch has been set up to assist in matters connected with violations of Commission regulations and licenses.

(4) In line with the Commission's increased emphasis on development of regulations and guides, the Acting Director of Regulation has appointed a Rules Clearance Committee reporting directly to him. This will be a standing committee on the development of regulations and will be composed of representatives of the Division of Licensing and Regulation, Division of Compliance and the Office of Radiation Standards. The General Manager also will designate a representative to attend meetings of the Commission to keep his office informed of proposed regulations.

(L. A. Mann)

Action on Reactor Projects by Licensing and Regulating Bodies

The AEC is required by the Atomic Energy Act to protect the public from undue hazards resulting from exposure to radioactivity. To fulfill this responsibility while furthering its simultaneous responsibility for developing the use of nuclear energy, the Commission has enacted and amended regulations and has published them in the *Code of Federal Regulations*. On Apr. 8, 1961, the Commission published in the *Federal Register* some proposed amendments to Title 10 of the Code, Part 50, Licensing of Production and Utilization Facilities, to specify conditions under which a licensee will have freedom of judgment in determining whether a contemplated change in facility equipment or operation contains an element of hazard not previously reviewed and approved by the Commission.[35] Comments by interested parties were invited. The proposed amendments were based on a Nov. 2, 1960, "Memorandum and Order" to the General Electric Company regarding the VBWR.

As of Mar. 13, 1961, the facilities license application record[36] was that reported in Table VI-2. In addition, 39 export licenses had been issued and 4 were pending.

The status of several power and test reactors in various stages of study, construction, and operation is shown in Table VI-3. Some of the more interesting features of the status of reactors and facilities are summarized below.

Big Rock Point Reactor (Docket 50-155)

The status of the 240-Mw(t) boiling-water reactor to be located at Big Rock Point, Mich., was reviewed in the June 1961 issue of *Nuclear Safety*.[37] On Apr. 27, 1961, the applicant, Consumers Power Company, again advised[38] that it will postpone requesting another hearing until the test program on Dresden type control-rod drives and blades (see below) is completed and that it may wish to initiate a new proceeding to obtain approval of technical specifications.

Dresden Reactor (Docket 50-10)

The Dresden reactor, a 630-Mw(t) boiling-water reactor, was shut down Nov. 15, 1960, when trouble with a control rod became apparent after a comparatively trouble-free period of operation.[39,40] The initial investigation revealed cracked control-rod-drive tubes, and, as mentioned previously,[37] the operator (Commonwealth Edison Company) proposed modifications to the control-rod drives and requested AEC review of the proposals. They also requested

Table VI-2 STATUS OF LICENSE APPLICATIONS AS OF MAR. 13, 1961

Type of reactor or facility	Number of licenses in status shown				
	Construction permits		Applications withdrawn	Operating licenses	
	Pending	Issued		Issued	Terminated
Power	2	7		3	
Test		1	2	2	
Research		17	31	67	5
Critical experiment facilities		3	1	14	1
Production facilities	3	1			

Table VI-3 CALENDAR OF LEGAL STEPS IN LICENSING U. S. POWER REACTORS

REACTOR INFORMATION					CONSTRUCTION PERMIT					OPERATING LICENSE				CURRENT STATUS
CODE NO.	NAME OWNER AND OPERATOR LOCATION	REACTOR TYPE	MAXIMUM POWER Mwt	MAXIMUM POWER Mwe	ACRS APPROVAL	HEARING EXAMINER APPROVAL	CONSTRUCTION PERMIT NO.	CONSTRUCTION PERMIT DATE	SCHEDULED COMPLETION DATE	ACRS APPROVAL	HEARING EXAMINER APPROVAL	OPERATING LICENSE NO.	OPERATING LICENSE DATE	
50-155	**BIG ROCK POINT** Consumers Power Co. Charlevoix County, Michigan	Boiling Water	240	75	Mar. 1960	May 1960			1962					
PP-4	**BONUS** USAEC Puerto Rico Water Resources Authority Rincon, Puerto Rico	Boiling Water Nuclear Superheat	50	16.3	Mar. 1960	June 1960	Not Required		Sept. 1962					
None	**BORAX-V** USAEC Argonne National Laboratory NRTS, Idaho	Experimental Boiling Water, Nuclear Superheat	35.7	2.7			Not Required					Not Required		Startup planned in 1961
50-144	**CVTR** Carolinas-Virginia Nuclear Power Associates Parr, South Carolina	Pressurized D_2O Moderator and Coolant	60.5	17	Dec. 1959	Apr. 1960	CPPR-7	5-4-60	Dec. 1962					
50-10	**DRESDEN** Commonwealth Edison Co. Grundy County, Illinois	Boiling Water	630	192 gr 180 net			CPPR-2	5-4-56	Construction started 3-57; completed Sept. 1959	May 1959	Sept. 1959 May 1961	DPR-2	9-26-59 11-16-59 6-2-60 10-14-60 6-9-61	On line 6-5-61 after being shut down 11-15-60
PP-1	**ELK RIVER** USAEC Elk River Rural Cooperative Power Association Elk River, Minnesota	Boiling Water Thorium Converter	73	22	Aug. 1959	Dec. 1959	Not Required		Jan. 1962					
None	**EBWR** USAEC Argonne National Laboratory Lemont, Illinois	Experimental Boiling Water	100	4.5			Not Required					Not Required		Operating since Dec. 1956
None	**EBR-II** USAEC Argonne National Laboratory NRTS, Idaho	Experimental Breeder Fast Neutron Spectrum Sodium Cooled	62.5	16.5			Not Required					Not Required		
None	**EGCR** USAEC Tennessee Valley Authority Oak Ridge, Tennessee	Experimental Graphite Moderated Helium Cooled	84	22	May 1959		Not Required		Dec. 1962					
F-16	**FERMI** Power Reactor Development Co. Detroit Edison Co. Monroe County, Michigan	Fast Neutron Spectrum Plutonium Breeder Sodium Cooled	300	104 gr 94 net			CPPR-4	8-4-56 12-10-58 5-26-59	Dec. 1961?					On 6-12-61 the United States Supreme Court upheld validity of construction permit
PP-3	**HALLAM** USAEC Consumers Public Power District Hallam, Nebraska	Graphite Moderated Sodium Cooled	240	75	July 1959 Feb. 1960	June 1960	Not Required		July 1962					
50-133	**HUMBOLDT BAY** Pacific Gas & Electric Co. Buhne Point, California	Boiling Water	163	60	June 1960 July 1960	Oct. 1960	CPPR-10	11-9-60	1962					
	IMPROVED CYCLE BOILING WATER REACTOR PROTOTYPE													Approved as basis for AEC contract negotiations, under

50-133	**HUMBOLDT BAY** Pacific Gas & Electric Co. Buhne Point, California	Boiling Water	163	60	June 1960 July 1960	Oct. 1960	CPPR-10	11-9-60	1962					
	IMPROVED CYCLE BOILING WATER REACTOR PROTOTYPE USAEC Cities of Los Angeles and Pasadena Departments of Water and Power Los Angeles, California	Boiling Water		50					1964					Approved as basis for AEC contract negotiations under Power Demonstration Reactor Program
50-3	**INDIAN POINT (CETR)** Consolidated Edison Co. of New York Buchanan, New York	Pressurized Water Thorium Converter	585	275 gr 255 net		Final Design June 1961	CPPR-1	5-4-56	Dec. 1961					Hearing for provisional operating license scheduled for Dec. 1961 or early 1962
None	**N. S. SAVANNAH** USAEC States Marine Lines Constructed at Camden, New Jersey	Pressurized Water	69	22,000 shp			Not Required		Dec. 1960					In July 1961, AEC authorized startup and operation through initial sea trials
	ORGANIC COOLED REACTOR PROTOTYPE USAEC Undetermined Utility Undetermined Location	Organic Cooled		50					1964					
50-130	**PATHFINDER** Northern States Power Co. Sioux Falls, South Dakota	Boiling Water Nuclear Superheat	203	66 gr 62 net	Feb. 1960	Apr. 1960	CPPR-8	5-12-60	July 1962					
50-171	**PEACH BOTTOM** Philadelphia Electric Co. York County, Pennsylvania	High Temperature Unclad Fuel Graphite Moderated Gas Cooled	115	40					1963					ACRS report on provisional construction permit expected late in 1961 or early in 1962
PP-2	**PIQUA OMR** USAEC (Reactor) City of Piqua Municipal Power Commission Piqua, Ohio	Organic Moderated Organic Cooled	45.5	11.4	July 1959	Jan. 1960			Oct. 1961	May 1961				Provisional operating license hearing late in 1961 or early in 1962
None	**PWR** USAEC Duquesne Light Co. Shippingport, Pennsylvania	Pressurized Water	231	68 gr 60 net	Jan. 1954					Nov. 1957				Operating since Dec. 1957
50-146	**SAXTON** Saxton Nuclear Experimental Corporation Berks County, Pennsylvania	Pressurized Water	20	5	Sept. 1959	Jan. 1960	CPPR-6	2-11-60	Jan. 1962	July 1961	Oct. 1961			ACRS approved startup program including operation up to full power
None	**SRE** USAEC Atomics International & Southern California Edison Santa Susana, California	Graphite Moderated Sodium Cooled	20	6										Operating since Apr. 1957
	SOUTHERN CALIFORNIA EDISON Southern California Location	Pressurized Water		360					1965					
50-18	**VBWR** General Electric Co. Alameda County, California	Test Reactor Boiling Water	40	5			CPPR-3	5-14-56				DPR-1	8-31-57	Operating since Sept. 1957
50-183	**VESR** General Electric Co. Alameda County, California	Light Water Moderated, Separate Steam Superheater	12.5		May 1961	July 1961		8-10-61						
50-29	**YANKEE (YAEC)** Yankee Atomic Electric Co. Rowe, Massachusetts	Pressurized Water	392	145 gr 136 net	Sept. 1957		CPPR-5	11-4-57		Feb. 1960 May 1960 June 1960	April 1960 July 1960 June 1961	DPR-3	7-9-60 7-29-60 6-23-61	On 6-23-61 Yankee's Provisional License was converted to a 40-yr Operating License

AEC concurrence with resumption of operation after successful conclusion of a test program.[41] A hearing was set for Mar. 28, 1961.

Upon further examination, however, it was discovered that some cracking of the boron steel control-rod blades had occurred. The operator submitted therefore, on February 27, proposed control-rod blade revisions.[42] At the same time the operator submitted plans for a test program for studying a new type of fuel (type II, stainless-steel-clad) along with used type I elements.[42] These proposals were considered at a meeting[43] of the ACRS and the AEC staff on Mar. 2–4, 1961, and on March 16 the Division of Licensing and Regulation requested further information on the control-rod system.[44]

At hearings concluded on March 29, Hearing Examiner Jensch ordered that the operator be allowed to resume operations under limited specified conditions at powers up to 100 kw(t), and he scheduled another hearing for April 10 for further presentation of evidence and cross examination.[45] On March 31, authorization was granted for limited test operation under the specified conditions.[46] The ACRS had notified the Commission by letter of March 4 that it believed the proposed limited operation would be safe.[47]

By letter dated April 8, the ACRS advised the Commission[48] that it believed the use of the new B_4C-filled control rods would be safe in operation at full power if a specified reinspection program were followed after 1500, 3000, 4500, and 9000 hr of operation but that it was not yet convinced that the new rods would be suitable for the life of the reactor. It also recommended occasional integral leak tests of the containment.

At hearings conducted on April 17, the integrity of the lower grid support was discussed because some cracks were discovered.[49] Discovery of these cracks had caused postponement of fuel loading. Further hearings held the last week of April included a discussion of possible consequences of the cracks in the support grid and a discussion of control-rod handling.[50]

Elk River Reactor (Docket PP-1)

The Elk River 58-Mw(t) boiling-water reactor was discussed by the applicant (Rural Cooperative Power Associates), the ACRS, and representatives of the AEC Divisions of Reactor Development and Licensing and Regulation on Mar. 2–4, 1961. The ACRS reported[51] to the Commission that the additions and corrections to the Hazards Summary Report with regard to control-rod drives, control rods, and fuel-element design did not require their further review, but that they would like to be kept informed of AEC staff action. The report stated that "upon completion of the staff action, the Committee believes that this reactor can be operated without undue risk to the health and safety of the public." A hearing was held on March 7–9 by the AEC staff, but action was postponed until further evidence is available for presentation.[52] On May 8, the Division of Licensing and Regulation requested specific information from Allis-Chalmers Manufacturing Company regarding cracks observed in the Elk River pressure vessel, including plans for remedial action.[53]

Enrico Fermi Fast-Breeder Reactor (Docket F-16)

The Enrico Fermi reactor, a 300-Mw(t) fast-breeder sodium-cooled reactor, is under construction at Lagoona Beach, Mich., by the Power Reactor Development Company. The U. S. Court of Appeals for the District of Columbia, in a 2 to 1 decision, on June 10, 1960, ordered that the construction permit granted by the AEC be set aside. The AEC and the applicant appealed the decision to the U. S. Supreme Court.[54]

Presentation of briefs to the Supreme Court began in the week of Feb. 13, 1961. The Court of Appeals had held that the AEC must make the same safety findings in granting a construction permit as in granting an operating license, that the AEC's safety findings were ambiguous, and that the AEC had not made a finding of compelling reasons for locating the reactor near a heavily populated area.[54] The applicant and the AEC presented arguments that the language and actions of the AEC had been correct and that the decision of the Court of Appeals was counter to the Atomic Energy Act. Briefs were filed by the International Union of Electrical, Radio, and Machine Workers in the week of Mar. 27, 1961, asking the Supreme Court to uphold the decision of the lower court[55] and claiming that the issuance of the construction permit by AEC, with the particular language used, was a violation of the Act and AEC regulations.

Questions set by the Court for argument were (1) whether the Act requires the Commission to make the same findings with regard to the safety

of operation before granting a construction permit as are required before granting an operating license and (2) whether the Act precludes the Commission from approving a site for a power reactor close to a populated area unless the Commission finds compelling reasons for such a location.[56] The attorneys presented extensive and opposing views in arguments before the Court, beginning in the last week of April.

On June 12 the Supreme Court reversed the judgment of the lower court in a 7 to 2 decision upholding the legality of the AEC order granting a provisional construction permit to the applicant.[57] The majority opinion supported the Commission and the applicant on both counts set for argument by the Court.

FWCNG Reactor (Docket 50-152)

Work on a proposed 153-Mw(t) D_2O-moderated gas-cooled reactor was terminated[58] by the AEC in early May 1961. The Commission has proposed that the applicants, the East Central Nuclear Group and the Florida West Coast Nuclear Group, consider continuing exploration of natural-uranium systems of mutual interest. The project had been oriented to the development of beryllium-clad fuel elements since August 1960, and the applicants had hoped to submit a revised proposal by Jan. 1, 1961, looking toward construction of an operating plant. The applicants concluded, however, that technical and economic uncertainties made it unfeasible to commit themselves on plant construction without further research and development. It had been hoped that a 50-Mw(e) plant would be constructed in Polk County, Florida.

Indian Point Reactor (Docket 50-3)

A 585-Mw(t) pressurized-water thorium-converter reactor is under construction for the Consolidated Edison Company of New York, Inc. On February 21 the applicant submitted an amendment to its application that described proposed designs for access hatches, for leakage monitoring of the containment vessel, and for other monitoring and instrumentation.[59] On March 2 the AEC forwarded to the Company a revised schedule of regulatory actions regarding the facility license application,[60] which provided for initial criticality in November of 1961.

On March 4, the ACRS reported favorably on the safety aspects of the reactor.[61] It was noted in the report that the applicant had ordered 1100°F heat-treated 17-4-PH stainless-steel control-rod mechanism parts in order to avoid the difficulty experienced with the Dresden control-rod drives (see Dresden reactor, above). The Committee expressed doubt that full-power operation can be attained without in-core monitors. The applicant was subsequently requested to submit additional information on several specific items which could involve changes in core reactivity when the reactor is shut down[62] and on components fabricated of precipitation-hardened stainless steel (particularly 17-4-PH stainless steel).[63]

On May 9 and 12, hearings were held for approval of the final design of the facility, with the approval to relate only to design and construction.[64] The technical specifications were to be taken up at a later date. The six Company and two AEC witnesses presented favorable testimony. The safety aspects of the facility were discussed in considerable detail, particularly by R. H. Bryan of the Research and Power Branch of the AEC Division of Licensing and Regulation, who explained the Commission's favorable report. He testified to the effect that, with appropriate administration and satisfactory resolution of the control-rod-drive mechanism problem, the Division believes that the facility can be operated without undue hazard to the health and safety of the public.

Lockheed Radiation Effects Reactor (RER) (Docket 50-172)

The ACRS reported to the Commission that after again reviewing the Lockheed 10-Mw(t) reactor on Mar. 2–4, 1961, it saw no reason to change its opinion that continued operation of the reactor at 10-Mw(t) in its present design form can be justified only for work essential to the national defense.[65,66] The ACRS concluded, however, that the reactor can be operated without undue risk to the public "provided (1) the power is limited to one megawatt thermal . . . , and (2) the excess reactivity be limited to the minimum required for operation at this stated power level. It is, of course, assumed that the present procedural safeguards and environmental surveillance will continue."

Nuclear Merchant Ship N. S. Savannah

The N.S. *Savannah* is to be powered by a 60-Mw(t) pressurized-water reactor. Public hear-

ings on its operation were held[67] the week of March 6. In testimony at the hearings, the AEC Maritime Administration Group presented an introduction and outline of the project; three contractors (George C. Sharp, Babcock & Wilcox, and New York Shipbuilding) discussed specific aspects of the program, including ship design and marine accidents, nuclear power plant and reactor hazards, construction and design of the containment vessel and shielding, and the test program to be followed in the sea trials. Review and inspection programs were described by representatives of the Oak Ridge National Laboratory, American Bureau of Shipping, U.S. Coast Guard, Public Health Service, the AEC Division of Compliance, and the AEC Division of Licensing and Regulation. Representatives from California and New York requested copies of all hazards reports[67] and subsequent test reports. It was reported by the representative from New York that neither the state nor the City objects to the N.S. *Savannah* entering the Port of New York. The status of control-rod drives made of 17-4-PH steel and control blades made of boron steel was discussed. Witnesses differed on the need for more tests on the drives, with the AEC witness testifying that more tests should be made. Testimony of the witnesses agreed that the stainless-steel-clad control blades appear to be satisfactory. The New York Shipbuilding Company witness, M. M. Mann, described the five-phase testing program, which extends through the Phase V Sea Trials, and discussed the possible return of the N.S. *Savannah* to Camden after the sea trials.

Maritime Administration officials have been working on obtaining approvals for the entry of the N.S. *Savannah* into foreign ports.[68] It was reported that no country has turned down the entry of the vessel. The Maritime Administration believes that third-party liability questions will be cleared up in ample time for such visits.

The hearings of the week of April 10 were chiefly related to the status of the test program and the use of 17-4-PH steel in the control-rod-drive mechanisms.[69] It was testified by AEC witness E. G. Case that the proposed program of tests and inspection of specified 17-4-PH components is acceptable. The program was outlined and explained by Maritime Administration and AEC witnesses. The New York Shipbuilding Company described the status of the ship propulsion plant tests, reporting that they were ready to proceed with reactor systems coordinated tests (dummy run).

Plum Brook (NASA) Reactor Facility (Docket 50-30)

The Plum Brook facility includes a 60-Mw(t) light-water-cooled and -moderated reactor with a primary beryllium reflector and a secondary water reflector. It is primarily for use in the development of "nuclear propulsion and nuclear-power generation." On Mar. 3, 1961, Examiner Bond recommended the granting of an 18-month provisional operating license to NASA[70] for thermal powers up to 100 kw, with specified restrictions. The license was granted[71] Mar. 14, 1961.

Saxton Reactor (Docket 50-146)

The Saxton facility includes a 20-Mw(t) pressurized-water reactor. On Apr. 20, 1961, the applicant, Saxton Experimental Corporation, requested issuance of a provisional operating license for a schedule of increasing operating powers[72] up to 20 Mw(t). A final safeguards report was submitted with the application.

Vallecitos Test Reactor (Docket 50-18)

On Mar. 2, 1961, the AEC Division of Licensing and Regulation ordered that the boiling-water test reactor at Vallecitos not be restarted pending review of tests and data on 17-4-PH steel control-rod-drive parts[73] (see Dresden reactor, above). On Apr. 13, 1961, the above order was vacated and permission to resume operation was granted under specified conditions relating to control-rod testing, inspection, and behavior.[74,75]

Vallecitos Experimental Superheat Reactor (Docket 50-183)

The General Electric Company applied Feb. 1, 1961, for a construction permit and 10-year license to operate the Vallecitos Experimental Superheat Reactor (VESR).[76] The VESR is designed as a research and development facility, largely for testing reactor superheater fuel. It is planned for ultimate operation in conjunction with the VBWR, reviewed earlier, with a nominal maximum power of 12.5 Mw(t). Completion of construction was predicted for 1962. Amendments[77,78] were submitted March 14 and April 11

regarding operation in conjunction with the VBWR. Technical information on safety features was presented.

Yankee Reactor (Docket 50-29)

On March 31 the Yankee Atomic Electric Company submitted information on operation of the Yankee reactor at 392 Mw(t), the design power level, to substantiate its claim that operation at powers up to 485 Mw(t) can be accomplished within limits imposed by the technical specifications.[79] The Company requested authorization to operate at power levels up to 485 Mw(t) for the full term of their license, which is scheduled to expire Nov. 4, 1997. Hearings were to be scheduled both on the proposition of obtaining a permanent license and for approval of a 485-Mw(t) operating level. Yankee requested that the two hearings be combined and hoped the amended license could be issued by the middle of June. On April 26 the request for a combined hearing was granted,[80] and June 8 was set as the date for the hearing.[81]

Proposed 200-Mw(t) Organic-Cooled Reactor

In a letter dated Apr. 10, 1961, the ACRS reported to the Commission its conclusion that each of six sites proposed for a 200-Mw(t) organic-cooled power-demonstration-program reactor is acceptable.[82] The proposals were submitted by the following utilities:

1. Dairyland Power Cooperative, Inc., LaCrosse, Wis.
2. Plains Electric Generation and Transmission Cooperative, Inc., Albuquerque, N. Mex.
3. Grand River Dam Authority, Vinita, Okla. (two sites).
4. Burlington Electric Light Department, Burlington, Vt. (two sites). (L. A. Mann)

Safeguards Reports and Selected Reading

The recently issued safeguards reports and selected literature pertaining to hazards of reactors are listed below for reference. Because of the similarity of many reactors (in particular, research reactors), this list is not intended to be all inclusive.

1. J. L. Murray, Preliminary Hazards Summary Report for the Vallecitos Superheat Reactor, USAEC Report GEAP-3643, General Electric Co., Atomic Power Equipment Dept., Feb. 1, 1961.

2. G. L. Blackshaw and C. H. Skeen, Safeguards Summary for the AE-6 Reactor, USAEC Report NAA-SR-5499, Atomics International, Mar. 15, 1961.

3. F. J. Remick and N. J. Palladino, Hazards Summary for the Nuclear Research Reactor Located at the Curtiss-Wright Nuclear Research Laboratory of the Pennsylvania State University, Report NP-9360, Curtiss-Wright Nuclear Research Laboratory and Pennsylvania State University, September 1960.

4. *Improving the AEC Regulatory Process*, Vols. I and II, U. S. Government Printing Office, Washington, D. C., 1961.

5. W. H. Berman and L. M. Hydeman, *The Atomic Energy Commission and Regulating Nuclear Facilities*, University of Michigan Law School, Ann Arbor, 1961.

6. C. G. Bigelow et al., Preliminary Hazards Report for a Reactor Experiment at NRTS, July 1960, USAEC Report PWAC-324, Pratt & Whitney Aircraft Div., United Aircraft Corp., Apr. 1, 1961. (Classified)

7. C. J. DeBevec, Amendment No. 1 to Preliminary Hazards Report for University of Wisconsin Pool Type Nuclear Reactor, Report NP-9361, General Electric Co., Atomic Power Equipment Dept., July 27, 1960.

8. R. E. Durand et al., SRE Standard Operating Procedures, USAEC Report NAA-SR-Memo-5326, Atomics International, June 27, 1960.

9. T. L. Gershun (Ed.), Preliminary Safeguards Report Based on Uranium-Molybdenum Fuel for the Hallam Nuclear Power Facility, USAEC Report NAA-SR-3379, Atomics International, Aug. 2, 1960.

10. S. E. Beall et al., Molten-Salt Reactor Experiment Preliminary Hazards Report, USAEC Report CF-61-2-46, Oak Ridge National Laboratory, Feb. 28, 1961.

11. Puerto Rico Water Resources Authority and General Nuclear Engineering Corp., Boiling Nuclear Superheater (BONUS) Power Station Preliminary Design Study and Hazards Summary Report, USAEC Report TID-8524, Vols. 1–4, June 1960.

12. Yankee Atomic Electric Co., Technical Information and Final Hazards Summary Report. Part B, License Application, AEC Docket No. 50-29, USAEC Report TID-6318 (Vols. I and II), 1960.

13. D. Gibbons et al., Radioactivation Analysis—A Bibliography, First Supplement, British Report AERE-I/R-2208(Suppl. 1), October 1960.

14. National Naval Medical Center, Hazards Summary Report for the Defense Atomic Support Agency Biomedical Radiation Research Facility, Addendum I, Report NP-9388, Sept. 20, 1960.

15. Consumers Power Company, Application to U. S. Atomic Energy Commission for Reactor Construction Permit and Operating License, Report NP-9359, 1960.

16. R. A. Egen et al., Hazards Summary Report for the VMR Critical Assembly Experiments, USAEC Report BMI-1445, Battelle Memorial Institute, June 10, 1960.

17. J. W. Chastain et al., Addendum to Hazards Summary Report for the GCRE Critical-Assembly Experiments, USAEC Report BMI-1379(Rev.), Battelle Memorial Institute, Nov. 14, 1960.

18. J. D. Simpson, Summary Hazards Evaluation of the SMR Facility, USAEC Report APEX-536, General Electric Co., Aircraft Nuclear Propulsion Dept., December 1959.

19. S. S. Rosen (Ed.), Hazards Summary Report for the Army Package Power Reactor SM-1, Task XVII, USAEC Report APAE-2(Rev. 1), Alco Products, Inc., May 1960.

20. J. M. Jacobs (Comp.), Selected Reactors of the Power Reactor Demonstration Program: A Literature Search, USAEC Report TID-3556, June 1960.

21. R. H. Clark and T. C. Engelder, Spectral Shift Control Reactor Basic Physics Program Critical Experiment Hazard Evaluation, USAEC Report BAW-1211, Babcock and Wilcox Co., September 1960.

22. R. J. Smith (Comp.), Reactor Safety: A Literature Search, USAEC Report TID-3525 (Rev. 2), September 1960.

23. J. R. Horan (Ed.), Annual Report of the Health and Safety Division, 1958, USAEC Report IDO-12012, Idaho Operations Office, 1958.

24. D. A. Dingee and J. W. Chastain, Jr., Hazards Summary Reports for the Battelle Plastic Reactor Facility, Report BMI-ACRS-659, Battelle Memorial Institute, Jan. 20, 1961.

25. B. Yates (Comp.), Bibliography on Nuclear Propulsion for Ships, British Report DEG-Inf. Ser.-242, 1960.

26. Union Carbide Nuclear Co., Final Hazards Summary Report for the UCNC Research Reactor, Report NP-9674, November 1960.

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31. C. C. Bigelow et al., Preliminary Hazards Report for a Reactor Experiment at CANEL, USAEC Report PWAC-314, Pratt & Whitney Aircraft Div., United Aircraft Corp., Jan. 20, 1961. (Classified)

32. E. R. Christie and B. W. Mar, Hazards Summary Report: The Kukla Prompt Critical Assembly, USAEC Report UCRL-6105, University of California Lawrence Radiation Laboratory, Feb. 24, 1960.

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34. Philadelphia Electric Co., Before the USAEC, Application of Philadelphia Electric Co. for Construction Permit and Class 104 License. Part B, Preliminary Hazards Summary Report, Peach Bottom Atomic Power Station, Vol. I, Plant Description and Safeguards Analysis, Report NP-9115, July 1960.

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2. General Electric Co., Vallecitos Boiling Water Reactor, Semiannual Report on Operating Experience Pertinent to Safety, No. 2, March 1960.
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4. L. Kornblith and R. H. Leyse, Report on VBWR, *Power Eng.*, 64(5): 41–44 (May 1960).

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